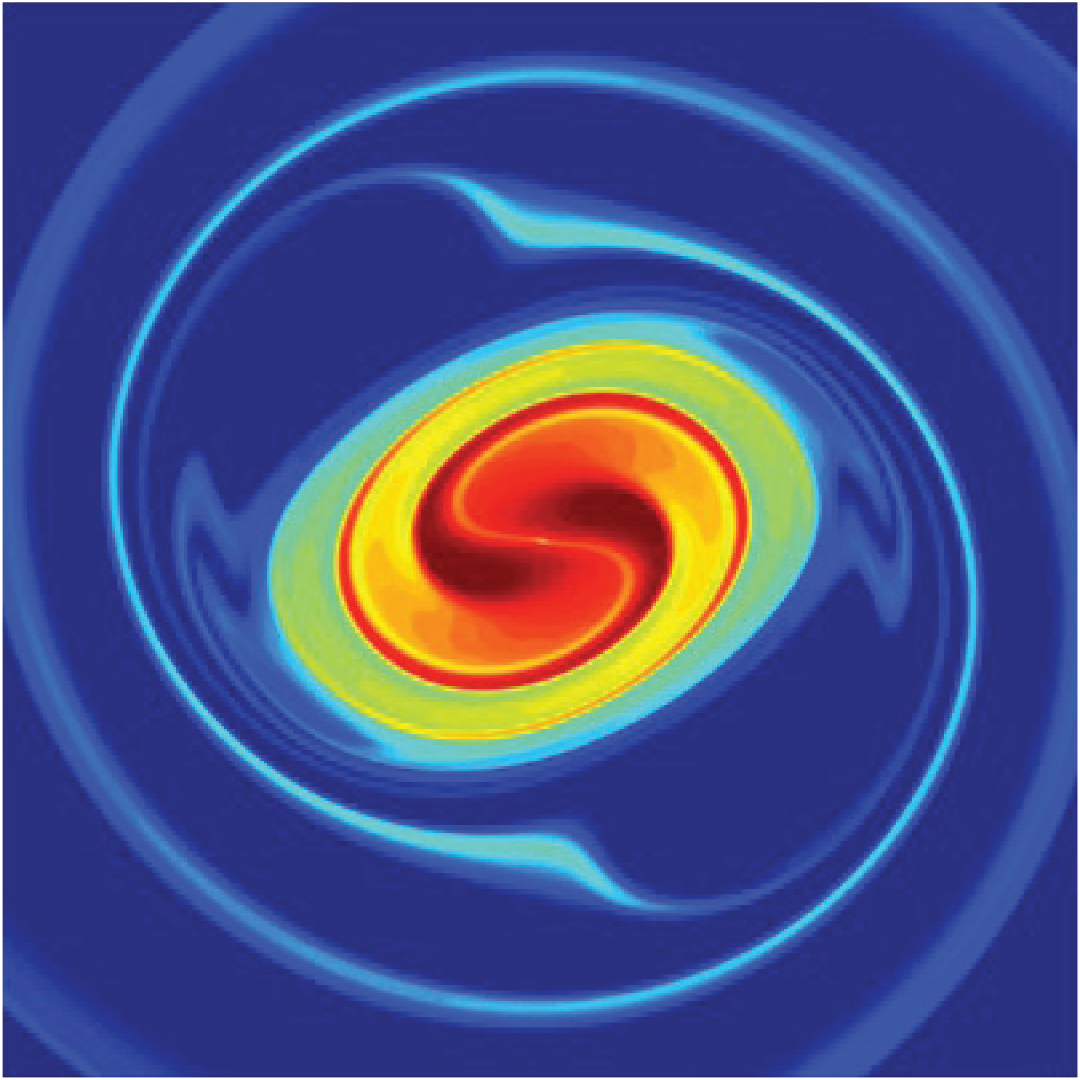
Research Needs for Magnetic Fusion Energy Sciences



Report of the Research Needs Workshop (ReNeW)
Bethesda, Maryland – June 8-12, 2009

U.S. DEPARTMENT OF ENERGY | Office of Science
OFFICE OF FUSION ENERGY SCIENCES

ON THE COVER

Depicted is Vortex Waltz, a computer simulation snapshot of two-dimensional fluid vortexes captured by J. Luc Peterson (graduate student) and Greg Hammett (physicist) at the Princeton Plasma Physics Laboratory, Princeton University. Two-dimensional fluid vortexes attract, swirling and merging with their partners in a turbulent ballet. This natural behavior influences phenomena ranging from weather patterns in the atmosphere to the performance of nuclear fusion devices. Advanced numerical algorithms and high-performance supercomputers allow for turbulence simulations of unprecedented detail. This snapshot catches the vortexes in the act. Originally entirely separated, the two vortex centers (dark red) have sent out spiral bands and shock waves throughout the background fluid as they've circled each other and combined. If left alone long enough, the two will complete their dance as a single, larger vortex. The image is featured in the 2009 Art of Science exhibit at Princeton University.

Research Needs for Magnetic Fusion Energy Sciences

Report of the Research Needs Workshop (ReNeW)

ReNeW is a planning activity of the Office of Fusion Energy Sciences (OFES)

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**On detail to OFES from Princeton Plasma Physics Laboratory.*

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EXECUTIVE SUMMARY

Nuclear fusion — the process that powers the sun — offers an environmentally benign, intrinsically safe energy source with an abundant supply of low-cost fuel. It is the focus of an international research program, including the ITER fusion collaboration, which involves seven parties representing half the world's population. The realization of fusion power would change the economics and ecology of energy production as profoundly as petroleum exploitation did two centuries ago.

The 21st century finds fusion research in a transformed landscape. The worldwide fusion community broadly agrees that the science has advanced to the point where an aggressive action plan, aimed at the remaining barriers to practical fusion energy, is warranted. At the same time, and largely because of its scientific advance, the program faces new challenges; above all it is challenged to demonstrate the timeliness of its promised benefits.

In response to this changed landscape, the Office of Fusion Energy Sciences (OFES) in the US Department of Energy commissioned a number of community-based studies of the key scientific and technical foci of magnetic fusion research. The Research Needs Workshop (ReNeW) for Magnetic Fusion Energy Sciences is a capstone to these studies. In the context of magnetic fusion energy, ReNeW surveyed the issues identified in previous studies, and used them as a starting point to define and characterize the research activities that the advance of fusion as a practical energy source will require. Thus, ReNeW's task was to identify (1) the scientific and technological research frontiers of the fusion program, and, especially, (2) a set of activities that will most effectively advance those frontiers. (Note that ReNeW was not charged with developing a strategic plan or timeline for the implementation of fusion power.)

The Workshop Report

This Report presents a portfolio of research activities for US research in magnetic fusion for the next two decades. It is intended to provide a strategic framework for realizing practical fusion energy. The portfolio is the product of ten months of fusion-community study and discussion, culminating in a Workshop held in Bethesda, Maryland, from June 8 to June 12, 2009. The Workshop involved some 200 scientists from Universities, National Laboratories and private industry, including several scientists from outside the US.

Largely following the Basic Research Needs model established by the Office of Basic Energy Sciences (BES), the Report presents a collection of discrete research activities, here called "thrusts." Each thrust is based on an explicitly identified question, or coherent set of questions, on the frontier of fusion science. It presents a strategy to find the needed answers, combining the necessary intellectual and hardware tools, experimental facilities, and computational resources into an integrated, focused program. The thrusts should be viewed as building blocks for a fusion program plan whose overall structure will be developed by OFES, using whatever additional community input it requests.

Part I of the Report reviews the issues identified in previous fusion-community studies, which systematically identified the key research issues and described them in considerable detail. It then considers in some detail the scientific and technical means that can be used to address these issues. It ends by showing how these various research requirements are organized into a set of eighteen thrusts. Part II presents a detailed and self-contained discussion of each thrust, including the goals, required facilities and tools for each.

This Executive Summary focuses on a survey of the ReNeW thrusts. The following brief review of fusion science is intended to provide context for that survey. A more detailed discussion of fusion science can be found in an Appendix to this Summary, entitled "A Fusion Primer."

Fusion Science

Fusion's promise

The main advantages of producing power from fusion reactions are well known:

- Essentially inexhaustible, low-cost fuel, available worldwide.
- High energy-density of fuel, allowing straightforward base-load power production without major transportation costs.
- No production of greenhouse gas, soot or acid rain.
- No possibility of runaway reaction or meltdown that could pose a risk to public safety.
- Minimal proliferation risk.
- Only short-lived radioactive wastes.

Few of these benefits are unique to fusion; what is exceptional is their simultaneous achievement in a single concept. For example, fusion's freedom from greenhouse-gas production and chemical pollution is shared with, among other energy sources, fission nuclear power; in this regard the relatively mild radioactivity of fusion, whose waste is thousands of times less radioactive and long-lived than fission, is significant. On the other hand, compared to the non-proliferating renewable energy sources, fusion offers a steady, predictable energy source with low land use.

To be weighed against these advantages is the long and relatively expensive development path for fusion. Achieving the conditions necessary for appreciable fusion reactions to occur invokes substantial physics and engineering challenges. Yet the impressive progress achieved in addressing these hurdles must be acknowledged. One measure is the exponential increase in fusion power produced in laboratory experiments, amounting to some eight orders of magnitude (a factor of 100,000,000) since the mid-1970s. Indeed some fusion experiments have approached scientific "break-even," producing roughly as much fusion power as was externally supplied for heating the fuel. A more important if less easily measured avenue of progress lies in scientific understanding. Fusion scientists have developed a broad and sophisticated, if still incomplete, picture of what is happening in a magnetically confined fusion plasma. This advance now allows routine control of key plasma properties and behavior.

Magnetic confinement

Magnetic confinement (more accurately termed "magnetic insulation") allows the fusion fuel, which is necessarily in the form of ionized gas, or plasma, to retain sufficient heat to maintain fusion reactions. It acts by enforcing a relatively low plasma density at the plasma boundary, where vessel walls would otherwise cool the gas, and by inhibiting heat flow from the interior to the wall region. The essential ingredient is a magnetic geometry in which the magnetic field lines abide in a closed, bounded region.

During the last decades of the twentieth century, fusion research gained important scientific victories in plasma confinement: major advances in both the control of instability and the amelioration of heat transport. While significant confinement issues remain to be solved, and while most of the fusion scientific community looks forward to substantial further improvements, the present demonstrated level of confinement is sufficient to impart confidence in the future of fusion energy. One indicator of this scientific advance is the rapid confinement progress mentioned above. Perhaps a more significant consequence is the decision by the international fusion community to embark on the ITER project.

Breadth of fusion research

Fusion progress requires scientific research of the highest quality and originality. Such science is not an activity to be balanced against the energy goal, but rather an essential component of the quest for that goal. This Report emphasizes the goal-directed nature of the program, but it is also appropriate to mention that, like any deep investigation, fusion research has enjoyed broad connections with other domains of science.

Many connections are mentioned in the Theme chapters of Part I. Examples are:

- Gyrokinetic simulation, used to understand transport and stability in magnetized fusion plasmas, has become an important tool in astrophysics and magnetosphere physics.
- Magnetic reconnection, a key phenomenon in the stability of magnetically confined plasmas, has central importance in numerous solar, magnetosphere and astrophysical contexts.
- Turbulent heat transport across the magnetic field, which plays a role in modern fusion experiments very similar to its role in the equilibrium configuration of the sun and other stars.
- Unstable Alfvén waves, whose effects in fusion experiments are closely similar to observed perturbations in the earth's magnetosphere.
- The high-strength, ductile materials being developed for fusion should have wide application in industry, including aerospace and chemical manufacturing.

Research requirements

In the next two decades, the "ITER era," magnetic fusion will for the first time explore the burning plasma regime, where the plasma energy is sustained mostly by its own fusion reactions. We expect ITER to expand our understanding of fusion plasma science and to be a major step toward practical fusion energy. It will also, as the first burning plasma experiment, pose new requirements, including advanced diagnostics for measurement and control in a burning-plasma environment, and analytical tools for understanding the physics of self-heating.

To benefit fully from its investment in ITER the US must maintain a broad research program, attacking fusion's scientific and technical issues on several fronts. We need in particular to acquire knowledge that ITER cannot provide: how to control a burning plasma with high efficiency for indefinite periods of time; how to keep a continuously burning plasma from damaging its surrounding walls — and the walls from contaminating the plasma; how to extract the fusion energy from a burning plasma efficiently and use it to produce electricity and a sustained supply of tritium fuel; and ultimately how to design economical fusion power plants. These requirements motivate a multi-disciplinary research program spanning such diverse fields as plasma physics and material science, and advancing a range of technologies including plasma diagnostics, magnets, radiofrequency and microwave sources and systems, controls, and computer simulation.

The key scientific and technical research areas whose development would have a major effect on progress toward fusion energy production were systematically identified, categorized and described in the three resource documents that form the starting point for ReNeW: the report of the Priorities, Gaps and Opportunities Panel, chaired by Martin Greenwald; the report of the Toroidal Alternates Panel, chaired by David Hill; and the report of the Energy Policy Act Task Group of the US Burning Plasma Organization.

In Part I of the ReNeW Report the full panoply of fusion issues are summarized and then examined from the point of view of research requirements: the facilities, tools and research programs that are needed to address each. The research thrusts presented in Part II are essentially integrated combinations of these research requirements.

The ReNeW Thrusts: A Research Portfolio

Thrust definition

The ReNeW thrusts listed are the key results of the Workshop. They constitute eighteen concerted research actions to address the scientific and technological frontiers of fusion research. Each thrust attacks a related set of fusion science issues, using a combination of new and existing tools, in an integrated manner. In this sense each thrust attempts a certain stand-alone integrity.

Yet the thrusts are linked, both by scientific commonality and by mutual dependence. The most important linkages — for example, requirements that a certain thrust be pursued and at least in part accomplished before another is initiated — are discussed in Part II of the main Report. Here we emphasize that fusion advances along a broad scientific and technological front, in which each thrust plays an important role.

The thrusts span a wide range of sizes, from relatively focused activities to much larger, broadly encompassing efforts. This spectrum is expected to enhance the flexibility of OFES planning. ReNeW participants consider all the thrusts to be realistic: their objectives can be achieved if attacked with sufficient vigor and commitment. Three additional elements characterize, in varying degrees, the ReNeW thrusts:

- Advancement in fundamental science and technology — such as the development of broadly applicable theoretical and simulation tools, or frontier studies in materials physics.
- Confrontation with critical fusion challenges — such as plasma-wall interactions, or the control of transient plasma events.
- The potential for major transformation of the program — such as altering the vision of a future fusion reactor, or shortening the time scale for fusion's realization.

Thrust organization

The resource documents used by ReNeW organized the issues into five scientific and technical research areas. Correspondingly, the ReNeW organizational structure was based on five Themes, each being further sub-divided into three to seven panels. The thrusts range in content over all the issues delineated in the five Themes.

Many of the ReNeW thrusts address issues from more than one Theme. For this reason the scientists contributing to most thrusts are from a variety of research areas, and key elements of a given thrust may stem from ideas developed in several Themes. In other words, the content of a typical thrust transcends that of any single Theme. Nonetheless, it is convenient to classify each thrust according to the Theme that contains its most central issues.

The ReNeW Thrusts are:

THEME 1: BURNING PLASMAS IN ITER

ITER participation will be a major focus of US fusion research during the time period considered by ReNeW. The opportunities and challenges associated with the ITER project are treated in Theme 1.

Thrust 1: Develop measurement techniques to understand and control burning plasmas. This Thrust would develop new and improved diagnostic methods for measuring and controlling key aspects of burning plasmas. The desired measurement techniques must be robust in the hostile burning-plasma environment and provide reliable information for long time periods. While initially focused on providing critical measurements for ITER, measurement capability would also be developed for steady-state burning plasmas beyond ITER.

Thrust 2: Control transient events in burning plasmas. This Thrust would develop the scientific understanding and technical capability to predict and avoid disruptions and to mitigate their consequences, in particular for ITER. Also, tools would be developed to control edge plasma transport and stability, to minimize instability-driven heat impulses to the first wall.

Thrust 3: Understand the role of alpha particles in burning plasmas. Key actions would be developing diagnostics to measure alpha particle properties and alpha-induced fluctuations, incorporating validated theories for alpha particle behavior into integrated burning-plasma simulation tools, and expanding the operating regime of burning plasma devices through the development of control techniques for alpha-driven instabilities.

Thrust 4: Qualify operational scenarios and the supporting physics basis for ITER. This Thrust would address key issues in forming, heating, sustaining, and operating the high-temperature plasmas required for ITER's mission. An integrated research campaign would investigate burning-plasma-relevant conditions with the use of upgraded tools for heating and current drive, particle control and fueling, and heat flux mitigation on existing tokamaks, along with a possible new facility.

THEME 2: CREATING PREDICTABLE, HIGH-PERFORMANCE, STEADY-STATE PLASMAS

An economic fusion reactor will require a steady state with higher fusion density and greater fraction of self-heating than ITER. This Theme addresses a broad range of issues, including both plasma physics and engineering science, needed to demonstrate that plasmas with the needed conditions can be achieved and controlled. Predictive capability to enable confident extrapolation to a demonstration reactor is emphasized.

Thrust 5: Expand the limits for controlling and sustaining fusion plasmas. This Thrust would integrate development of the diagnostic, auxiliary heating, current drive, fueling systems and control systems needed to maintain the nonlinear tokamak plasma state, seeking to maximize performance. The Thrust will exploit existing experiments to test and develop new ideas and proceed with increased integration in upcoming steady-state experiments and alpha-heated plasmas in ITER, ultimately enabling the self-heated and self-driven plasmas needed for a fusion power plant.

Thrust 6: Develop predictive models for fusion plasmas, supported by theory and challenged with experimental measurement. Advances in plasma theory and simulation would be combined with innovative diagnostic methods and experiments to improve and validate models of confined plasma dynamics. Assessment of critical model elements would be provided by dedicated analysts, acting as bridges between theorists, code developers and experimentalists.

Thrust 7: Exploit high-temperature superconductors and other magnet innovations to advance fusion research. Magnets are crucial for all MFE concepts. This focused Thrust would perform the research necessary to enable revolutionary new high-temperature superconducting materials to be used in fusion applications. Key activities include development of high-current conductors and cables, and integration into components of fusion research experiments, with great potential to improve their design options.

Thrust 8: Understand the highly integrated dynamics of dominantly self-heated and self-sustained burning plasmas. This Thrust would explore scenarios where, as in a reactor, most heat comes from fusion alphas and most current is self-driven by plasma gradients. It would start by assessing potential advanced plasma scenarios and upgrades on ITER which could enhance its performance. In parallel, scoping/design studies would be done for a new US facility to explore the high fusion gain DEMO plasma regime. The studies would support actions to proceed with ITER enhancements, the construction of a US deuterium-tritium (D-T) facility, or both.

THEME 3: TAMING THE PLASMA-MATERIAL INTERFACE

Magnetic confinement sharply reduces the contact between the plasma and the vessel walls, but such contact cannot be entirely eliminated. Advanced wall materials and magnetic field structures that can prevent both rapid wall erosion and plasma contamination are studied in Theme 3.

Thrust 9: Unfold the physics of boundary layer plasmas. Comprehensive new diagnostics would be deployed in present confinement devices to measure key plasma parameters in the boundary region, including densities and temperatures, radiation, flow speeds, electric fields and turbulence levels. The results could vastly improve numerical simulation of the edge region, allowing, in particular, reliable prediction of wall erosion and better radiofrequency antenna design.

Thrust 10: Decode and advance the science and technology of plasma-surface interactions. Measurement of complex interaction of plasma with material surfaces under precisely controlled and well-diagnosed conditions would provide the information needed to develop comprehensive models to uncover the basic physics. These measurements would be made on both upgraded present facilities and new boundary plasma simulators capable of testing irradiated and toxic materials.

Thrust 11: Improve power handling through engineering innovation. Heat removal capability would be advanced by innovative refractory power-exhaust components, in parallel with assessment of alternative liquid-metal schemes. Materials research would provide ductile, reduced-activation refractory alloys, which would be developed into prototypes for qualification in high-heat flux test devices. Practical components would be deployed on existing or new fusion facilities.

Thrust 12: Demonstrate an integrated solution for plasma-material interfaces compatible with an optimized core plasma. Understanding of interactions between a fusion plasma core region and its boundary would be advanced and validated in a new facility. The facility would combine high power density, long pulse length, elevated wall temperature and flexibility regarding boundary systems, in a limited-activation environment. Knowledge gained from Thrusts 9-11 would help guide the design of this facility.

THEME 4: HARNESSING FUSION POWER

Fusion energy from D-T reactions appears in the form of very energetic neutrons. Theme 4 is concerned with the means of capturing this energy, while simultaneously breeding the tritium atoms needed to maintain the reaction.

Thrust 13: Establish the science and technology for fusion power extraction and tritium sustainability. Fusion must create the tritium fuel it uses, and do so in the same systems that capture and extract the fusion energy. This Thrust develops the scientific foundation and engineering of practical, safe and reliable processes and components that harvest the heat, create and extract the tritium, and rapidly process and contain the tritium. The Thrust will culminate in a fuel and power handling capability on a scale needed for a demonstration energy system.

Thrust 14: Develop the material science and technology needed to harness fusion power. The objective of this Thrust is to create low-activation, high-performance materials that effectively function for a long time in the hostile fusion environment. An essential requirement to fulfill the mission of this Thrust is the establishment of a fusion-relevant neutron source to perform accelerated characterization of the effects of radiation damage to materials.

Thrust 15: Create integrated designs and models for attractive fusion power systems. Advanced design studies, focused primarily on DEMO, but also on nearer term fusion nuclear facilities, is one element of this Thrust. These would lay out the scientific basis for fusion power and provide focus to the research efforts required to close the knowledge gap to DEMO. The other element comprises science-based predictive modeling capabilities for plasma chamber components and related systems.

THEME 5: OPTIMIZING THE MAGNETIC CONFIGURATION

Currently most large fusion experimental devices are based on the tokamak magnetic configuration, a design using a strong, axisymmetric external magnetic field to achieve operating parameters close to those in a fusion reactor. Alternative magnetic configurations are studied to investigate physics and technology principles that could optimize the design of future fusion devices. The most developed alternate toroidal magnetic configurations are considered in Theme 5.

Thrust 16: Develop the spherical torus to advance fusion nuclear science. Experiments on the small aspect ratio tokamak, or Spherical Torus, would be extended to regimes of lower collision frequency, approaching values needed for fusion nuclear science applications. Plasma startup, power handling, controlled stability, and sustainment issues in this regime would be studied in long-pulse experiments using stronger magnetic fields, improved heating and current drive, and advanced diagnostics, with strong coupling to theory and modeling.

Thrust 17: Optimize steady-state, disruption-free toroidal confinement using 3-D magnetic shaping, and emphasizing quasi-symmetry principles. Magnetic quasi-symmetry in 3-D configurations is expected to lead to excellent plasma confinement while ensuring stable steady-state burning plasma performance with minimal need for control. This Thrust would conduct new quasi-symmetric experiments, which would, together with theory, engineering design, and targeted international collaboration, validate extrapolation to burning plasma applications.

Thrust 18: Achieve high-performance toroidal confinement using minimal externally applied magnetic field. This Thrust advances a multi-faceted program of theory, simulation, and well-diagnosed experiments to resolve critical issues of confinement, stability, and current sustainment in magnetic configurations with minimal toroidal field. New devices with heating and current drive systems would enable scaling to high temperature and small ion gyroradius. Fusion system studies will guide productive directions for present and future research.

Appendix: A Fusion Primer

Just as the heaviest elements, such as uranium, release energy when fission allows them to become smaller, so the very lightest elements release energy when they fuse, joining together to produce larger nuclei. (The dividing line between nuclei that are too light and want to fuse and those that are too heavy occurs at iron, the most stable nucleus.) The reaction that occurs most readily is the fusion of two isotopes of hydrogen: deuterium (D), whose nucleus consists of a proton and a neutron, and tritium (T), whose nucleus contains a proton and two neutrons. Fusion of these nuclei — the so-called D-T reaction — yields helium, an inert, non-radioactive gas whose nucleus has two protons and two neutrons. This helium nucleus or "alpha particle" carries 20% of the fusion energy production. It is contained by magnetic fields, and provides the plasma self-heating that sustains the very high plasma temperature. The remaining neutron is released at very high energy — energy whose capture provides 80% of the energetic profit of the reaction.

A reactor based on D-T reactions would have to breed tritium from lithium (which is plentiful), using the neutrons liberated in the D-T fusion process. More advanced fuel cycles would not require tritium breeding, but the D-T reaction has advantages with regard to accessibility and energy production. It is expected to be used in at least the first generation of fusion power reactors. Because all nuclei are positively charged, they electrically repel each other. This "Coulomb repulsion" can be overcome only by bringing the reactants to very high temperatures; in the case of D-T the required temperature exceeds one hundred million degrees.

Far below thermonuclear temperatures the electron on each hydrogen atom breaks free from its nucleus, yielding independent ion and electron fluids. The resulting electrically active gas, called *plasma*, can carry enormous electric currents; it is strongly responsive to electromagnetic fields, while at the same time able to produce strong fields on its own. Thus the operating fluid in any fusion device is plasma, a form of matter more electrodynamically active than any conventional liquid, solid or gas.

In summary, the key features of D-T fusion are:

1. An operating temperature in the hundred-million degree range, with the result that the working gas is necessarily in the plasma state.
2. An energy release primarily in the form of very fast alpha particles and neutrons, whose energy must be captured to provide the thermal output of the reactor.
3. The need to breed tritium from the D-T neutron and lithium.

Heating and confinement

Evidently the most basic tasks in constructing a fusion reactor are to heat a hydrogen gas to thermonuclear temperatures, and then to confine the resulting plasma for a time long enough for fusion reactions to take place, thus maintaining the high temperature. In most reactor designs heating is provided by a combination of driving electric currents through the plasma, directing energetic particle beams at the plasma, and energizing plasma particles by means of radiofrequency electromagnetic radiation, similar to the heating mechanism of a microwave oven.

Confinement is measured by the so-called energy confinement time, denoted by τ_E. Since both reaction rates and energy loss rates depend upon the plasma density *n*, the required value of τ_E depends on plasma density. It turns out that the critical parameter is the product $n\tau_E$; when density is measured in ions per cubic centimeter and τ_E in seconds, sufficient confinement has been achieved if the product exceeds about 10^{14} sec/cm^3 (the "Lawson criterion").

One way to satisfy the Lawson criterion is to compress a hydrogen pellet to extreme density values, exceeding the density of conventional solids, while allowing relatively short confinement times. This is the approach taken by the *inertial confinement* program. The main arm of international fusion research uses much lower densities — lower even than the density of air at the earth's surface. Thus the working fluid is a rarefied plasma, whose low density is part of the reason for the intrinsic safety of the device. The relatively long confinement time thereby required is supplied by magnetic fields, taking advantage of the plasma's strong response to such fields. This line of research is called *magnetic fusion*, although the phrase "magnetic confinement for fusion" would be more descriptive.

Magnetic confinement

Neon signs confine cold plasma in glass tubes. But a very hot, rarefied plasma — a fusion plasma — could not maintain thermonuclear temperatures if it had substantial contact with a material wall. At the densities used in magnetic fusion, plasma resting against a wall will quickly cool, bringing fusion reactions to a halt. So the confining magnetic field must protect the plasma from being quenched by contact with its bounding vessel. A magnetic field configured to provide this confinement is traditionally called a "magnetic bottle."

A magnetic bottle can work because charged particles — the ions and electrons that constitute a fusion plasma — spiral around the local field direction in helical orbits; the stronger the field, the tighter the helix. Thus, while motion parallel to the field is unaffected, motion perpendicular to the local field direction is strongly inhibited.

This inhibition of perpendicular motion has two effects. First, it allows the magnetic force to act against plasma pressure, pushing plasma away from the vessel wall. This profile control is especially effective when a *divertor* — a magnetic geometry in which the outermost field lines are diverted into an external chamber — is employed. In this case the layer of plasma near the vessel wall has especially low density, imposing a near vacuum between the inner plasma core and the wall.

The second insulating effect of the magnetic field pertains to dissipative transport. The inhibition of perpendicular motion affects plasma diffusion and heat conduction: transport in directions transverse to the field is sharply reduced, while transport parallel to the field is unaffected. For an appropriate field configuration this anisotropy markedly slows the conduction of heat from the fusion plasma core to the boundary region. Notice that this effect acts throughout the plasma volume, not only near the wall.

It is significant that while a magnetic bottle can reduce plasma contact with material boundaries, such contact is not eliminated. The residual contact is sufficiently tenuous to maintain a hot plasma interior, but still problematic because the wall material can be scarred. Aside from the obvious lifetime aspects of such erosion, plasma-wall interaction can allow impurities from the wall to enter the confinement region, with deleterious effects on both confinement and fusion reaction rates. Thus, significant materials-physics issues arise in the fusion quest.

A centuries-old theorem in topology shows that any closed surface on which the magnetic field does not vanish must have the topology of a torus: a magnetic bottle must be *toroidal* — donut-shaped. All the devices considered by ReNeW resemble donuts in this sense. (So-called "magnetic mirrors" get around the topological theorem by "plugging" the ends of a cylindrical field configuration; the mirror approach to confinement was not part of the purview of this ReNeW.) Since the only source of a magnetic field is electric current, magnetic confinement is based on electric currents flowing around or within some toroidal surface.

Most confinement devices employ a combination of external currents, in wire-wound coils, and internal currents, flowing within the plasma itself, to maintain the toroidal field structure. A prominent example is the *tokamak*, in which external and internal currents combine to yield a confining field that is symmetric with respect to a central axis. Other confinement schemes have yet to achieve the tokamak's level of performance but could bring operating advantages. For example, the stellarator deliberately breaks the field symmetry in order to simplify steady-state operation. And there are schemes under investigation that require relatively weak (and therefore less expensive) external magnetic fields.

Constructing a magnetic bottle does not solve the problem of confinement; there are essentially two additional hurdles. First, plasma currents, arising spontaneously from electromagnetic and fluid instability, can create magnetic fields that open up the bottle. Second, even when the magnetic configuration is stable with regard to gross distortion, localized "micro-instabilities" can produce fluctuations that degrade confinement. Common versions of such accelerated transport resemble boiling water on a stove: the water remains in the pot, but its turbulent motion rapidly conducts heat from the hot bottom to the cooler upper surface.

In the last decades of the twentieth century fusion research gained important scientific victories in plasma confinement: major advances in both the control of instability and the amelioration of turbulent transport. While significant confinement issues remain to be resolved, and while the fusion scientific community looks forward to substantial further improvements, the present demonstrated level of confinement is sufficient to impart confidence in the future of magnetic fusion energy.

Heating and confinement are the central, but not the only, challenges that must be faced before fusion power can be realized. Even a perfectly confined plasma at thermonuclear temperature must be fueled with reactant, it must be promptly cleansed of the helium that fusion produces, its thermal energy yield must be effectively retrieved, and so on. Such challenges occupy increasing research attention as the fusion program matures; they are the subject of major attention by ReNeW.

PART I:

ISSUES AND RESEARCH REQUIREMENTS

Part I: Issues and Research Requirements

Introduction

The central task of the ReNeW activity was to provide a strategic framework for realizing practical fusion energy. Beginning with the set of frontier issues identified in its resource documents, ReNeW first mapped the issues onto *research requirements* – the activities and tools needed to address the issues. Next, ReNeW developed a discrete set of research activities, here called "thrusts," each of which is a technically coherent set of actions designed to address research requirements and resolve issues. Thus ReNeW used a two-step map, first from issues to research requirements, and then from the latter to thrusts.

Resource documents → Research requirements → Thrusts

Part I of the Report is mainly devoted to the first step in this map, extracting the research requirements from the issues. It ends with a brief presentation of the second step, reserving a full description of the thrusts to Part II.

For completeness we begin by listing the ReNeW resource documents.

Resource documents

The set of key issues and knowledge gaps defining the fusion research frontier had been identified, characterized and prioritized in community-based reports prior to ReNeW. Three recent and especially pertinent reports were selected by ReNeW as its primary resource documents:

- The report of the Priorities, Gaps and Opportunities Panel. Chaired by Martin Greenwald, this panel identified the scientific and technical issues arising on a path to a demonstration fusion power plant, assuming the success of the ITER experiment.
- The report of the Toroidal Alternates Panel. Chaired by David Hill, this panel identified objectives and issues of a set of non-tokamak approaches to magnetic fusion.
- The report of the Energy Policy Act Task Group of the US Burning Plasma Organization. This report focused on objectives and issues associated with the US participation in the ITER experiment.

The lists of issues and gaps compiled in these reports were used by ReNeW as a starting point. While allowing for the possibility of modifying these lists, we avoided repeating the effort that had gone into making them.

ReNeW Organization: Themes

The structure of ReNeW was based on five themes, each comprising a major area of fusion research, and each being further organized into three to seven sub-panels. Three of the five themes were taken from the Greenwald Report. The remaining two themes focused respectively on ITER activities and alternate confinement concepts:

THEME 1: BURNING PLASMAS IN ITER

ITER participation will be a major focus of US fusion research during the time period considered by ReNeW. The many opportunities and challenges associated with the ITER project are treated in Theme 1.

THEME 2: CREATING PREDICTABLE, HIGH-PERFORMANCE, STEADY-STATE PLASMAS

The US has been a leader in demonstrating advanced plasma confinement scenarios that allow higher confinement efficiency, steady-state operation, and other advantages. Theme 2 addresses the science underlying such schemes.

THEME 3: TAMING THE PLASMA-MATERIAL INTERFACE

Magnetic confinement sharply reduces the contact between the plasma and the vessel walls, but such contact cannot be entirely eliminated. Advanced wall materials and magnetic field structures that can prevent both rapid wall erosion and plasma contamination are studied in Theme 3.

THEME 4: HARNESSING FUSION POWER

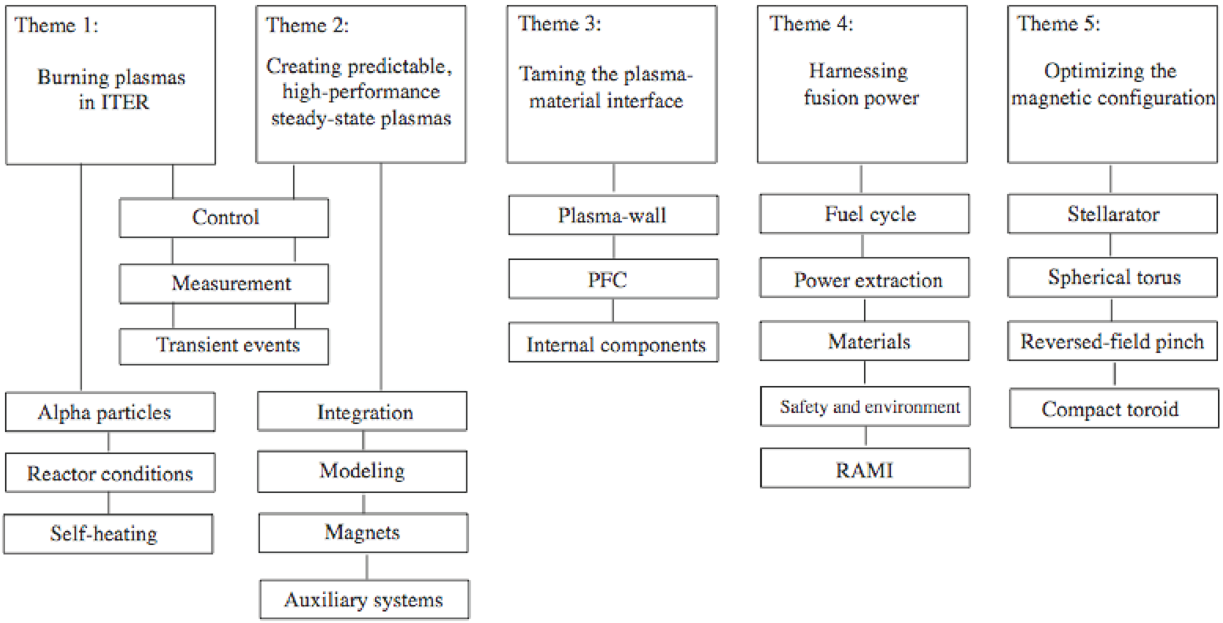
Fusion energy from deuterium-tritium (D-T) reactions appears in the form of very energetic neutrons. Theme 4 is concerned with the means of capturing this energy, while simultaneously breeding the tritium atoms needed to maintain the reaction.

THEME 5: OPTIMIZING THE MAGNETIC CONFIGURATION

Currently most large fusion experimental devices are based on the tokamak, a design using a strong, axisymmetric external magnetic field to achieve operating parameters close to those in a fusion reactor. Future fusion devices might also be tokamaks, but there are alternative design principles with potentially attractive features; the most interesting such designs are considered in Theme 5.

The themes in turn were organized into panels, each focused on certain issues of science and technology. The resulting structure is illustrated on the next page, with panel names abbreviated for convenience.

ReNeW Themes and Panels



Notice that three panels were shared between Themes 1 and 2, reflecting the particularly close relation between some of the issues explored in these two themes. More generally, the various themes explored many similar issues, and ReNeW took pains to insure regular and thorough communication among them.

Report Organization

Both parts of the Report are organized by themes. Thus, each chapter after this Introduction is devoted to one of the five themes, and issues reviewed and described in each chapter mirror the panels making up the appropriate theme. Each chapter in Part I begins by reviewing the issues and gaps within its purview, as specified in the resource documents. It next derives the research requirements pertinent to the particular theme, and then organizes these into thrusts. Each chapter ends by displaying the map between its particular set of issues and the corresponding thrusts.

The organization into theme Chapters is convenient, but potentially misleading: it might suggest that each thrust involved issues derived from a single theme. In fact, the thrust contents transcend the theme organization. Many thrusts combined issues from more than one theme, and the authors of a given thrust were in many cases gathered from the membership of various themes. Table 1 summarizes the relation between ReNeW themes and thrusts. It should be clear that all thrusts involve issues from more than one theme; in some cases the multiple-theme involvement pertains even to central concerns of the thrust.

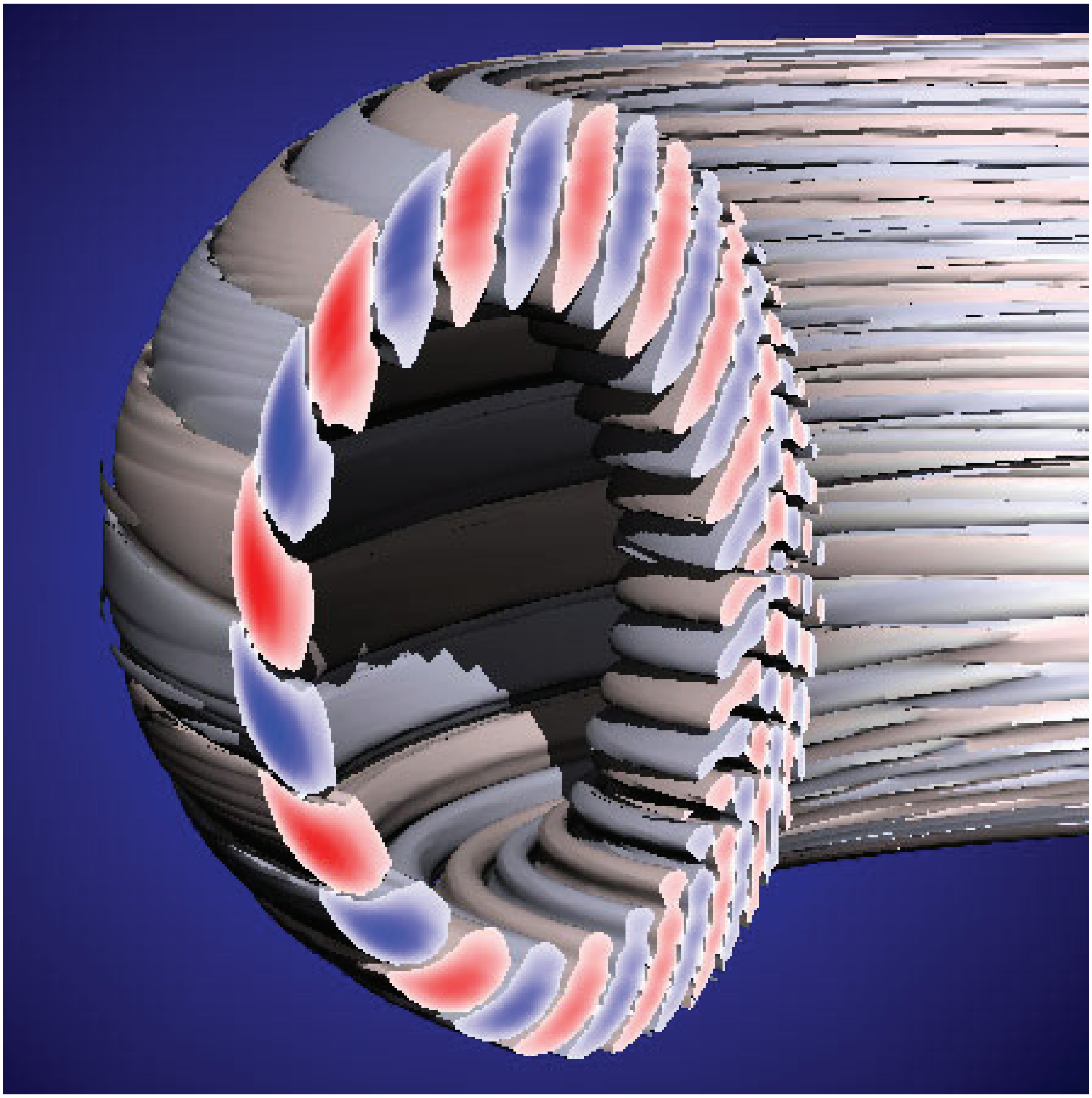
		Magnetic Fusion Energy Science (MFES) Research Requirements				
		Theme 1	Theme 2	Theme 3	Theme 4	Theme 5
		Burning Plasmas in ITER	Steady State High Performance	Plasma-Material Interface	Harnessing Fusion Power	Magnetic Configuration Optim.
Research Thrusts	1 Measurement	🟥	🟥	🟦	🟦	🟦
	2 Transient events	🟥	🟥	🟨	🟦	🟨
	3 Alpha particles	🟥	🟨	🟦	🟦	🟨
	4 ITER operational scenarios	🟥	🟦	🟦	🟦	🟦
	5 Control and sustainment	🟥	🟥	🟨	🟦	🟨
	6 Predictive models	🟥	🟥	🟥	🟦	🟥
	7 High temperature superconductors	🟦	🟥	🟨	🟨	🟥
	8 Integrated plasma dynamics	🟨	🟥	🟨	🟨	🟦
	9 Boundary layer plasma	🟥	🟥	🟥	🟦	🟥
	10 Plasma-material interactions	🟦	🟨	🟥	🟥	🟨
	11 Power handling innovation	🟦	🟨	🟥	🟥	🟨
	12 Core plasma - first wall integration	🟦	🟥	🟥	🟥	🟨
	13 Fusion power extraction and tritium	🟦	🟦	🟨	🟥	🟦
	14 Fusion materials	🟦	🟦	🟨	🟥	🟦
	15 Fusion power systems	🟦	🟦	🟨	🟥	🟦
	16 Spherical torus for fusion nuclear science	🟦	🟥	🟨	🟨	🟥
	17 3D magnetic shaping	🟦	🟥	🟦	🟦	🟥
	18 Minimal external magnetic field	🟦	🟨	🟦	🟦	🟥

🟥 Thrust activities are optimized or strongly focussed on Theme's Research Requirements
🟨 Thrust activities contribute significantly to Theme's Research Requirements
🟦 Thrust activities contribute marginally to Theme's Research Requirements

Table 1. ReNeW Thrusts Address Research Requirements. All MFES research requirements described in Part I are addressed by the set of Research Thrusts described in Part II. The Thrusts constitute the building blocks of a diverse, well-integrated MFES research program for the ITER era.

THEME 1:

BURNING PLASMAS IN ITER



ON PREVIOUS PAGE

Nonlinear simulation of a Toroidal Alfvén Eigenmode instability in a tokamak. (From D.A. Spong, ORNL)

THEME 1: BURNING PLASMAS IN ITER

Introduction

SCOPE AND FOCUS

The next frontier for magnetic confinement fusion science is the study of the burning plasma regime, in which the fusion process itself provides the dominant heat source for sustaining the plasma heat content (i.e., self-heating). The "burning plasma" regime for deuterium-tritium (D-T) plasmas is defined as Q≥5, where Q is the ratio of output fusion power to input heating power – a condition that implies at least 50% of the heating is provided by the ultra-energetic alpha particles created from D-T fusion reactions. It is anticipated that certain physics features of fusion plasmas in such self-heated conditions may be quite different from those encountered on present-day experiments in which the plasma is heated primarily by external means. Therefore, achieving and sustaining burning plasmas will have the dual benefit of proving the technical basis for fusion energy production, as well as improving the knowledge base about the burning plasma regime for optimization of future fusion devices.

No experiment has yet entered the burning plasma regime. During the 1990s, D-T experiments were performed in the Joint European Torus (JET) and the Tokamak Fusion Test Reactor (TFTR), which approached "break-even" (defined as fusion gain of Q=1). JET, for example, achieved 16 megawatts of fusion power and Q=0.6. TFTR carried out a campaign of 1,000 discharges of 50/50 D-T plasmas from December 1993 to April 1997.

ITER, an international project presently under construction, is designed with the specific purpose of achieving and studying burning plasmas (see Figure 1). The mission of ITER is to demonstrate the scientific and technical feasibility of fusion energy production for peaceful purposes. To meet this high-level programmatic objective, the ITER device will have technical performance objectives of Q=10 for 300-500 seconds in its baseline operation phase and Q=5 for 3000 seconds in its

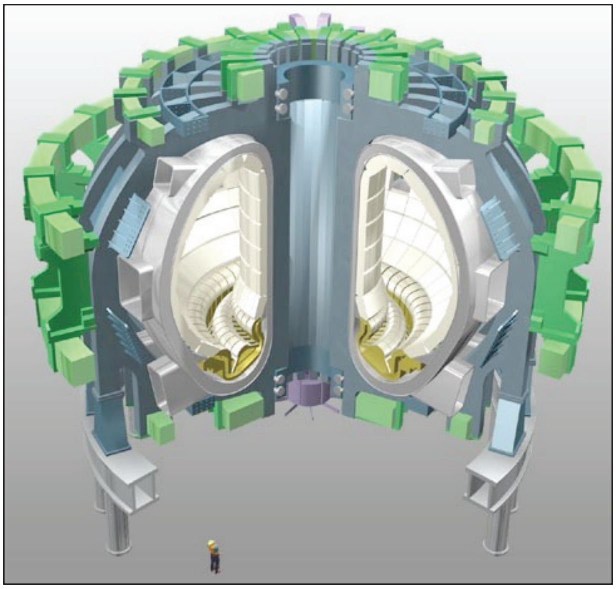


Figure 1. ITER burning plasma experimental facility, currently under construction in Cadarache, France, by a consortium of seven international partners, including the US.
(Figure courtesy of the ITER Organization.)

extended operation phase. The optimal result from the research on ITER will be to obtain a sufficient scientific and technical basis for operation at reactor-relevant engineering parameters and in burning plasma conditions and thus enable confidence in designing a next-step fusion device (e.g., DEMO).

Many of the scientific issues in burning plasmas — such as confinement, macro-stability, power and particle handling, long-pulse operation, diagnostics, and plasma control — are similar to those already being addressed in existing experimental facilities. What makes them more challenging are the scale-up to large reactor size and strong magnetic field; the need to operate near performance limits, with high heat flux on plasma facing components; and the thermonuclear environment, with harsh radiation and neutron fluxes, along with nuclear science issues such as tritium dust, tritium retention in the vessel walls, and the need to breed tritium fuel. In addition, as already mentioned, the distinguishing feature of a burning plasma is self-heating by a large population of alpha particles. The self-heating property means that the pressure, current, and rotation profiles in the plasma will be "autonomous" (i.e., self-organized), rather than controlled externally. The presence of the supra-thermal alpha particles, approximately 300 times more energetic than plasma ions and electrons, leads to new instabilities, anomalous transport, profile modification, perturbation of burn dynamics, and other impacts on the plasma behavior.

The topic of Theme 1 in this Research Needs Workshop was divided among the following six panels:

1. Understanding alpha particle effects.
2. Extending confinement to reactor conditions.
3. Creating a self-heated plasma.
4. Controlling and sustaining a self-heated plasma.
5. Mitigating transient events in a self-heated plasma.
6. Diagnosing a self-heated plasma.

Each of these panels contributed a section in this Chapter, in which the current status, research requirements, and scientific opportunities for the respective panel are described. The members of the panels, who represent a broad range of expertise in fusion physics and engineering science, are listed at the end of the chapter.

The scientific issues for achieving and understanding the burning plasma state in ITER, which is the overall topic of Theme 1, have a significant overlap with those for creating predictable, high-performance, steady-state plasmas, which is the topic of Theme 2. Thus, it was natural, in organizing the efforts of Themes 1 and 2, to set up some of the panels — viz., on plasma control (#4 in the list above), off-normal transient events (#5), and plasma diagnostics (#6) — as joint panels of Themes 1 and 2. However, the descriptions of the research issues and requirements in this chapter, even those for the joint panels, are focused on the research work needed to ensure that ITER will indeed be a success, whereas the Theme 2 chapter primarily looks ahead to DEMO and other steps after ITER. Nevertheless, it should be understood that topics of panels specific to Theme 2 — such as predictive modeling, heating and current drive systems, and plasma integration — are also of vital interest and importance to Theme 1. Topics of panels for Themes 3, 4, and 5 are also relevant to Theme 1.

A table is presented at the end of this chapter that shows how the research requirements of the six panels of Theme 1 map into high-priority Research Thrusts, to be described in Part II of this ReNeW Report.

The basic resource document for Theme 1 is *Planning for US Fusion Community Participation in the ITER Program* (also known as the "Energy Policy Act Report"). This is referenced at the end of this chapter, along with other suggestions for further reading.

Research Requirements

UNDERSTANDING ALPHA PARTICLE EFFECTS

Success in achieving and sustaining the burning plasma state depends on coupling as much of the energy of the D-T reactant alpha particles to the background plasma as possible. In a standard burning-plasma scenario, the energy of the fusion-produced alpha particles is transferred to the background plasma, and the alpha particles slow down due to electron drag. However, there is the possibility that alpha particles could be lost if their single-particle trajectories are unconfined or if they cause Alfvén waves to become unstable. Such losses are of concern not only due to the reduction in self-heating, but also because the unconfined high-energy particles could damage internal components of ITER.

CURRENT STATUS

Issues

Due to the high average energy of alpha particles, their properties (e.g., stability, transport, current drive) are theoretically expected to be substantially different from those of the background plasma ions and electrons. For this reason, there are many research issues that are specific to the alpha particle population itself (see Figure 2). These issues can be categorized as follows:

- How well are the alpha particles confined by the magnetic configuration, including 3-D fields?
- Do alpha particle populations that are consistent with operation at Q > 5 interact with the background plasma in a deleterious manner?
- Do the alpha particles couple to the dynamical behavior of the background plasma and cause a new set of instabilities and associated transport?
- Can the interaction of the alpha particle population with the background plasma be controlled for beneficial purposes?

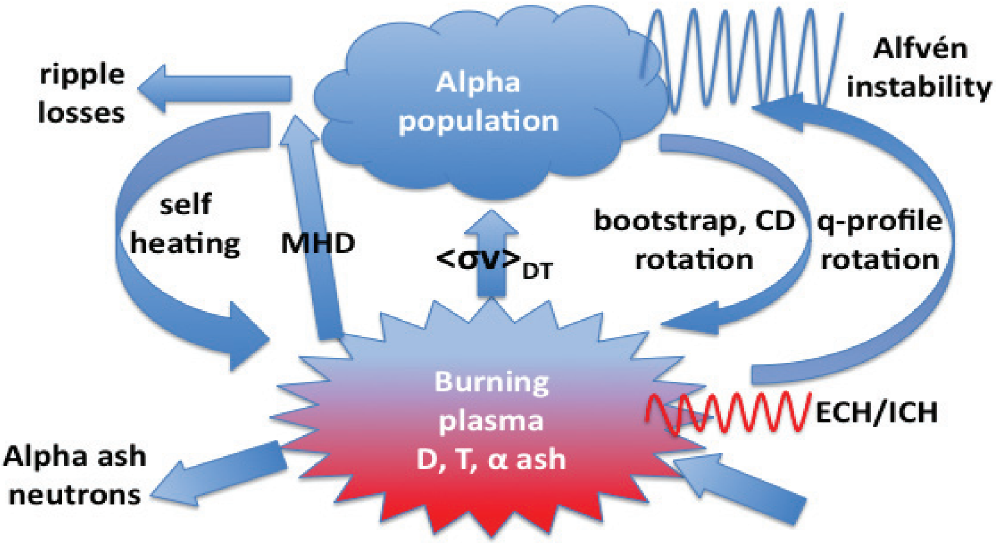


Figure 2. Alpha particles can cause new plasma phenomena (such as Alfvén instabilities), and their heating of the burning plasma drives new nonlinear feedback loops. (Figure courtesy of D.A. Spong.)

While many aspects of alpha particle physics have been anticipated and tested in existing experiments, it is not possible to produce fast ion populations that simultaneously achieve the parameters and velocity distribution that are characteristic of alpha particles in burning plasma devices such as ITER. Three such distinguishing features can be identified:

- Alpha particles in ITER are predicted to transit the device at rates comparable to the characteristic magnetohydrodynamics (MHD) wave frequency, leading to the possibility of resonant wave-particle interactions and instability drive.
- The spatial extent of the alpha particle gyro-orbit relative to the machine size is substantially smaller in ITER than in present-day devices, potentially leading to a larger number of unstable modes.
- Alpha particles in ITER are expected to have an isotropic velocity distribution, whereas fast ion populations in present-day devices typically are strongly anisotropic (and hence more unstable).

These intrinsic differences mean that only partial tests of fast ion instabilities and transport models are possible in present-day experiments; realistic tests will need to rely on power-plant scale experiments such as in ITER. Both validated predictive tools and advanced diagnostics on existing and future facilities are needed to bridge this parameter gap. Reliable simulations are also required to evaluate future control tools that will be needed to mitigate these instabilities and optimize the fusion burn in ITER and DEMO devices.

Understanding these issues will require a strong emphasis on advanced fast ion diagnostics and dedicated energetic particle/plasma experiments, together with improved theories and simulation codes. In parallel, neutron-hardened diagnostics for the ITER environment must be developed. The predictive alpha particle projects must be developed on a time scale compatible with their use in the full-machine simulation effort.

Recent accomplishments and progress

Considerable progress has been made in understanding alpha particle effects through closely coupled efforts in experiments, diagnostics, theory, and simulations. Highlights include:

- Alfvén instabilities excited by energetic ions have been theoretically predicted and observed in a range of magnetic configurations (e.g., small to large aspect ratio tokamaks, also stellarators).
- Linear Alfvén stability thresholds have been predicted and validated with diagnostics capable of measuring the internal spatial and spectral structure of multiple fluctuating fields and with external antennas capable of exciting stable Alfvén modes to measure their *in-situ* damping rates.
- Enhanced transport and direct loss of energetic ions associated with Alfvén instabilities have been observed.
- The potential for control of the fast ion instabilities has been demonstrated via localized radiofrequency wave heating (at both the ion and electron cyclotron frequencies).
- Fast ion instabilities with spatially localized, high-mode-number eigenmodes (similar to those expected in ITER) have been observed experimentally and modeled theoretically.
- Analysis of recently observed instabilities associated with acoustic (sound wave) coupling to Alfvén modes suggests a potential means for efficiently transferring alpha particle energy to the thermal plasma directly.

In addition, significant progress has been made in modeling energetic particle instabilities and their associated effects. New numerical codes based on extended-MHD theory are being developed for realistic nonlinear simulation of fast ion instabilities in burning plasmas. Already they have been used to successfully model various energetic particle-driven instabilities in existing experiments. Sophisticated gyrokinetic codes, with self-consistent treatment of the fast ion populations and also the thermal plasma fluctuations, are also under development and have been used to simulate Alfvén eigenmodes.

SCIENCE CHALLENGES, OPPORTUNITIES, AND RESEARCH NEEDS

The primary alpha physics issues revolve around energetic particle instabilities and their diagnosis, alpha confinement, and self-heating. Recent observations with new diagnostic techniques (see Figures 3 and 4) indicate that significant fast ion transport induced by multiple Alfvénic instabilities may be present in the core of reactor-relevant regimes. This underscores the urgent need to develop validated predictive models and effective control tools for Alfvénic instabilities. However, the existing theoretical, simulation, diagnostic, and control tools are not yet fully adequate to accurately predict the impact of fast ion-driven instabilities in present devices and in ITER. The research needs in the various research categories are listed.

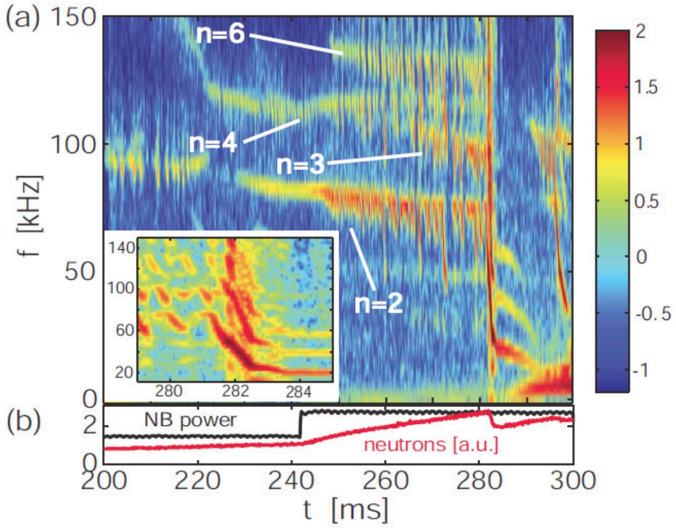


Figure 3. Spectrogram of the time-varying frequency of magnetic fluctuations showing the onset of toroidal Alfvén eigenmode activity after t = 220 ms, followed by an avalanche at t = 282 ms (detailed in the inset) in NSTX. (Figure reproduced from M. Podestà et al., Phys. Plasmas 16 [2009] 056104.)

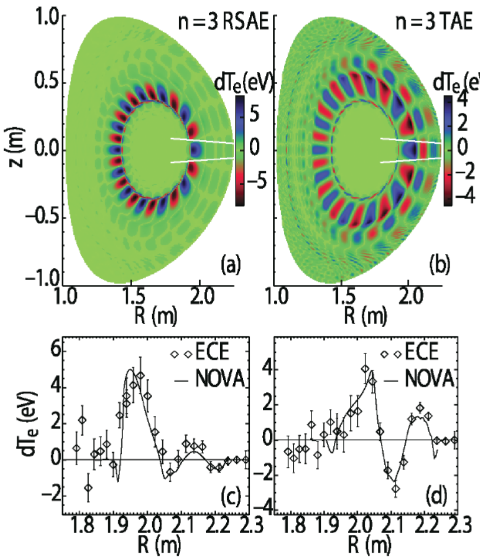


Figure 4. Radial structure of toroidal Alfvén eigenmodes measured in DIII-D with electron cyclotron emission diagnostic and compared to NOVA-K synthetic diagnostic simulation predictions. (Figure reproduced from M.A. Van Zeeland et al., Phys. Rev. Lett. 97 [2006] 135001.)

How well are the alpha particles confined by the magnetic configuration, including 3-D fields?

ITER and DEMO devices will have various sources of magnetic field non-axisymmetry, such as toroidal field ripple, ferromagnetic materials, blanket modules, control coils for edge localized modes and resistive wall modes, and internal MHD and Alfvén modes. Such asymmetries and time-varying fields can sensitively affect alpha particle confinement. Improved Monte Carlo simulation codes will need to be developed that take all of these effects into account. Although particle-following codes currently exist that can include a prescribed field of spatially non-axisymmetric and time-varying modes, the level of transport from such calculations often significantly underestimates the observed losses. Also, alpha particle confinement calculations in the future should be based on 3-D equilibrium calculations that include magnetic islands and self-consistently account for the effects of external ferromagnetic materials.

Existing methods for internal (confined) fast ion and lost particle measurements are inadequate to reconstruct a large fraction of the spatial and velocity space distribution of the fast ions. Also, present measurements cannot resolve the collective motion of the fast ions on the time scale of the wave period. These limitations must be overcome, both for confined and lost particle measurements. Resolving the phase space distribution of the energetic ions and their motion on the wave period time scale will lead to major advances in the understanding of the physics of anomalous fast ion transport in advanced confinement regimes.

Fast ion losses are currently measured on many experiments using scintillator detectors at a few locations just outside the plasma edge. However, since alpha particle losses in future burning plasmas will result in highly localized wall heat loads and longer-term helium-induced blistering, the loss patterns will need to be routinely evaluated with higher resolution than is currently available. Deployment of extensive arrays of fast ion detectors should be carried out on existing experiments to provide improved resolution maps of fast ion loss characteristics. These measurements could then be used to validate simulations of fast ion losses. Burning plasma relevant extensions of these fast ion loss techniques will need to be developed.

Do alpha particle populations that are consistent with operation at Q > 5 interact with the background plasma in a deleterious manner?

Present diagnostics of energetic particle-driven instabilities are limited to measurements of the internal density and temperature and the external magnetic field fluctuations. These provide only indirect indications of the instability mode structures and intrinsic wave-field amplitudes. Techniques are now available for direct measurements of the internal fluctuating electric (heavy ion beam probe) and magnetic fields (motional Stark effect and polarimetry). This information would substantially improve the validation of theoretical models, both for the growth and nonlinear saturation of the modes as well as prediction of the associated fast particle losses.

Over the time scale of ITER and DEMO, computational models of Alfvén stability (see Figure 5) will need to be made increasingly realistic through improved representations of geometry, inclusion of accurate finite-Larmor-radius effects in the alpha particle drive, accounting for radiation damping effects, and treating couplings between energetic particle modes and core plasma turbulence. The goals for such simulation efforts are expected to include: mapping out the stable landscape for burning plasma regimes; determining the accessible regions of phase space into which an unstable slowing-down distribution of alpha particles could be redistributed, while maintaining burning plasma conditions; and assessing whether unstable regimes exist with tolerable levels of alpha transport. In addition, such models will be essential for evaluating control methods for alpha-driven turbulence and in developing scenarios for alpha channeling.

Techniques for driving current through radiofrequency waves (such as lower hybrid current drive) involve anisotropic parallel energy transfer to electrons and will have some amount of parasitic absorption on alpha particles. While this can be kept small (~10%) through appropriate choice of heating frequency, what it does to alpha particle confinement requires further examination.

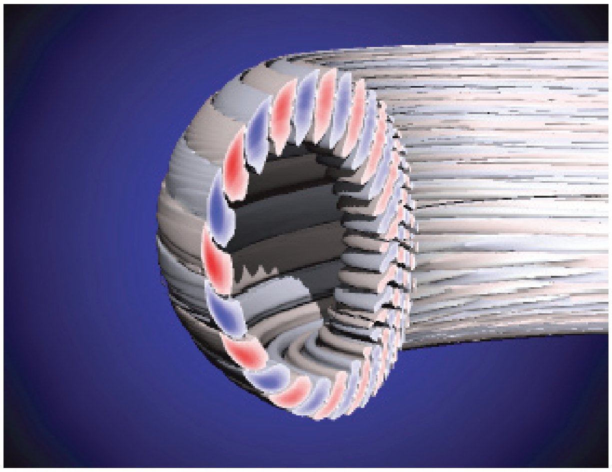


Figure 5. Nonlinear simulation of Alfvén instability in the DIII-D plasma (Figure from D.A. Spong and M.A. Van Zeeland, APS/DPP Bull. Am. Phys. Soc. 53 [2008] JP6.00091.)

Do the alpha particles couple to the dynamical behavior of the background plasma and cause a new set of instabilities and associated transport?

The strong self-heating of future burning plasmas will cause energetic particle behavior to be coupled with core stability and turbulence to a degree not observable on existing experiments. For example, it is expected that sawtooth crashes could redistribute fast ions and create conditions that are favorable to Alfvén “avalanche” modes. A related issue is the evolution of the core plasma into a high-pressure-gradient, energetic-particle-stabilized state that can rapidly become sawtooth/ballooning unstable as the stabilizing alphas are transported by energetic particle modes. At smaller scales, the question of how energetic particle instabilities interact with plasma microturbulence will become increasingly important in evaluating the damping rates of the shorter-wavelength Alfvén instabilities that will characterize ITER. The simultaneous investigation of the full spectrum of relevant instabilities and their interaction is complicated by the disparate time scales involved, from acoustic to ion cyclotron; new techniques will have to be developed to efficiently deal with this scale separation.

Can the interaction of the alpha particle population with the background plasma be controlled for beneficial purposes?

The successful operation of sustained flattop regimes in burning plasmas will require new techniques for active control of the alpha population. For example, methods for temporarily inducing enhanced alpha losses over limited energy ranges could provide instability suppression, burn control, or alpha ash removal. Such losses could be driven by external radiofrequency sources or time-varying ripple fields or by the destabilization of modes that resonantly transport alphas. Other control methods based upon rotation control, q-profile control, and use of radiofrequency and electron cyclotron resonance heating are expected to be useful for controlling alpha-driven fluctuation levels. Alpha channeling refers to the identification of mechanisms for directly transferring the alpha population energy to the deuterium and tritium ions without having to first pass through the electron channel. If successful, this could vastly improve the self-heating efficiency of the burning plasma. Other forms of channeling might involve using the alpha particles to assist with current drive, enhance plasma rotation, or reduce the alpha particle partial pressure.

Finally, it should be noted that existing alpha physics simulation tools have typically focused on time scales that are short compared to those that characterize core plasma transport and magnetic flux evolution. For the strongly self-heated regimes of ITER and DEMO, alpha physics and core plasma simulations must be fully integrated, using quasi-linear theory and other techniques, so that the impact of alpha transport and turbulence on the core plasma and *vice versa* can be self-consistently assessed. For example, the question of whether true steady-state profiles can be sustained in burning plasmas will be crucial for extrapolations to DEMO. Also, such a fully integrated model could be applied to better evaluate the stability of the alpha population, using realistic sources and sinks rather than arbitrarily imposed profiles and boundary conditions.

CONCLUSION

The goal of alpha physics research is to develop predictive understanding and control methods for strongly self-heated regimes in burning plasmas. This will involve making significant improvements in understanding alpha-driven Alfvénic instabilities through advanced integrated diagnostics, theory, and advances in simulation. These tools will need to be validated across a range of complementary existing and future fusion experimental devices.

EXTENDING CONFINEMENT TO REACTOR CONDITIONS

Adequate confinement of the plasma energy content is critical in achieving and sustaining the burning plasma state. In this regime, small changes in the confinement properties of the plasma can result in substantial changes in the degree of self-heating. Projections based on data from present-day devices indicate that energy confinement in ITER will be sufficient for achieving the burning plasma state. However, in terms of many important parameters, the ITER plasma conditions are a significant extrapolation from those obtained in present-day devices. To reduce the risk to the ITER mission introduced by these uncertainties, research is needed to characterize confinement properties in plasmas that closely match ITER conditions. Also, new techniques are needed for favorably modifying the confinement properties in ITER.

CURRENT STATUS

Issues

The observed transport of energy, particles, and momentum across the confining magnetic field in fusion-grade plasmas generally exhibits features that cannot be explained solely by collisional processes (as embodied in neoclassical transport theory). Turbulence resulting from wave-particle instabilities of very small spatial scale (comparable to the size of the gyro-orbits of the ions and electrons around the magnetic field) has been shown to be prevalent in these plasmas. Turbulent eddies that carry energy, particles, and momentum across the confining magnetic field are believed to play a key role in the observed level of transport. The severity of the turbulence-driven transport has been shown to be highly sensitive to the background plasma conditions and the energy flow through the confining magnetic field. In cases where there is significant local variation in the plasma flow or the pitch of the confining magnetic field, these turbulent eddies are sheared apart, leading to the formation of so-called “transport barriers” at these locations. Because of the sensitivity to plasma conditions, the boundary condition plays an important role in overall transport levels. This is best exemplified by a nearly doubled confinement improvement in plasmas that have a transport barrier very close to the plasma edge (the so-called H-mode [high-confinement mode] regime). The physics mechanisms behind the formation of this transport barrier and the subsequent structure of the kinetic profiles in this edge region (known as the H-mode pedestal) are not well understood in spite of intense research efforts.

Improving our understanding of energy, particle, and momentum confinement and determining how best to access and sustain H-mode operation are critical aspects of achieving the burning plasma regime in ITER. If the energy confinement is poorer than expected or the impurity buildup is greater than expected, then ITER may have difficulty in achieving its fusion performance goals. Hence, it is important to assess critical physics issues associated with confinement. These issues can be categorized in three broad areas:

- Improved characterization of confinement in plasma conditions specific to anticipated ITER operating scenarios.

- Improved understanding of the effect of non-axisymmetric fields on plasma confinement.
- Development of the capability to model transport and the conditions for transport barrier formation sufficiently well for simulating potential ITER scenarios.

The US fusion energy science community is a world leader in confinement and transport research, with flexible and comprehensively diagnosed experimental facilities (equipped with very good turbulence diagnostics), as well as excellent theory, modeling, and simulation capabilities. The field of confinement and transport provides a good mix of short and long-term research opportunities, both experimental and theoretical. Answering the questions that are motivated by the need to understand transport in burning plasmas prior to and during ITER operation will require a solid base program and ancillary facilities with the tools necessary to reproduce, study, and control plasma conditions as similar as possible to those of ITER. Given these capabilities, the US fusion energy science community will be well positioned to contribute strongly in this area.

Recent accomplishments and progress

Significant progress has been made toward a predictive understanding of energy, particle, and momentum transport, including detailed comparisons of experimental observations with theoretical predictions. Highlights include:

- Detailed measurements of fluctuations over a wide range of spatial scales.
- Agreement between experimental measurements of turbulence characteristics and kinetic profiles and predictions from gyrokinetic turbulence and gyro-Landau-fluid codes.
- Direct measurements of turbulence-induced zonal flows.
- Models developed that accurately predict the structure of the H-mode pedestal.
- Correlation found between the power threshold for low to high-confinement mode (L-H) transition and the edge plasma flow.
- Observation of self-generated plasma rotation in the absence of external torque.
- Clear evidence of inward convection of particle density.
- Identification of trigger mechanisms for internal transport barrier formation.
- Improved scalings with dimensionless parameters for projecting confinement to future devices.

SCIENCE CHALLENGES, OPPORTUNITIES, AND RESEARCH NEEDS

Improved characterization of confinement in plasma conditions specific to anticipated ITER operating scenarios

The burning plasma regime is expected to have specific characteristics. Simulations indicate that the optimum operating point for ITER has high density (~ 10^{20} m^{-3}), high temperature (> 15 keV), and strong magnetic field (~ 6 T). These requirements imply a specific range of values for dimensionless parameters of importance for transport. These values are somewhat different from those generally obtained in present-day devices. Particular examples include:

- The ion and electron characteristic gyro-orbit size relative to the machine size (known as the normalized gyroradius ρ*) is nearly an order of magnitude smaller in ITER than in present-day devices.
- The collision frequency of the ions and electrons in ITER relative to the frequency at which these particles transit a full circuit of the confining magnetic field (known as the collisionality ν*) is at the low end of the range of values obtained in present-day devices.
- The thermal equilibration time between ions and electrons in ITER is much smaller than the thermal confinement time, which implies close coupling between the ion and electron energy channels.

Because these parameters are important in the transport of energy, particles, and momentum, there is therefore still some uncertainty in projecting ITER confinement.

Energy Transport: Whereas the thermal transport of ions is thought to be reasonably well understood, additional work is needed to show that the present understanding is complete and correct. In particular, quantitative estimates of the role of rotation in suppressing turbulence and improving transport should be further refined. For electron thermal transport, there is no consensus on the identity of the underlying mechanism, which appears to be sensitive to physics related to spatial scales ranging from the ion to electron gyroradius. Diagnostic capability for both profile and fluctuation measurements needs to be expanded to solve this extremely demanding problem. Determining the scaling of transport phenomena with dimensionless parameters has proven to be a valuable tool in developing a better physics basis for extrapolation to ITER. A transport extrapolation to smaller normalized gyroradius (ρ*) and perhaps smaller collisionality (ν*), while keeping the other dimensionless variables fixed, is the preferred scaling path. Multi-machine studies of the ρ* and ν* dependences are needed to increase the range covered and thereby reduce projection uncertainties toward ITER.

Several enabling tools are required in this area. In particular, measurements capable of resolving fluctuations in the magnetic field, velocity, density, temperature, and electric potential on scale lengths comparable to the electron gyroradius (~ 5×10^{-3} cm in a typical fusion plasma) are needed to probe turbulence that drives electron transport. In addition, more electron heating than is currently available on existing machines will be needed to access electron-loss-dominated regimes. Such electron heating could also be used to better simulate the burning plasma regime with strong alpha particle heating of electrons and low torque injection.

Momentum Transport: The magnitude, direction, and detailed profile of the plasma flow are determined by the interplay of rotation sources, momentum transport, and any sink mechanisms that are present. Rotation is generated in a burning plasma (1) intrinsically or spontaneously, (2) externally by radiofrequency waves, (e.g., from the wave's momentum), and (3) through radiofrequency heating not externally driven. While extrapolation of intrinsic rotation from a multiple-machine database appears favorable (see Figure 6), it is important to determine the physical origin of spontaneous rotation and validate its size scaling. Pure electron heating cases are the most relevant to ITER. Recent observations of flow drive from mode conversion of radiofrequency waves look promising, although verification on a variety of devices, along with a more stringent comparison with theory, should be undertaken. Further work with heating from electron cyclotron waves and lower hybrid waves should be pursued as potential tools for profile control.

Particle Transport: The transport of particles (electrons, fuel ions, and impurity ions) has been experimentally shown to combine both turbulence-driven and collisional effects. The turbulence-driven transport is suspected to depend on particle charge and mass, whereas collisional effects are known to be strongly dependent on the charge of the particle. These differences in transport properties among particle species could be leveraged to isolate control of a particular species. To make firm predictions about density peaking and impurity accumulation, research should be targeted on micro-instabilities that are relevant to ITER. Future experiments should attempt to verify the pinch effects predicted by turbulence simulations and should include measurements of both majority and impurity ion particle transport as well as plasma fluctuations. A diagnostic for measuring electric potential fluctuations in the core of the plasma, such as a heavy ion beam probe, is especially desirable. Use of current drive to remove the effect of the Ware pinch (an inward drift of toroidally confined trapped particles) would make these studies cleaner.

Core fueling efficiency in ITER by gas puffing and pellets is critically dependent on both “normal” particle transport and transport of the pellet-sourced ions; in fact, pellet modification of the plasma profiles may affect the particle transport. Experiments need to be done to determine the effect of pellet injection angle and launch location and of the particle pinch on the core fueling efficiency. This includes experiments on high-field-side injection, simulations with realistic particle pinch, and loss due to pellet-induced edge localized modes. It is highly desirable to find some way to explore post-pellet transport in low-collisionality plasmas because the observed and predicted pinch is absent at high collisionality; perhaps frequent, very small pellets would be appropriate. Much of this work is already included on the lists of high-priority research for the International Tokamak Physics Activity (ITPA).

Pedestal Structure: Projections for ITER from models of turbulence-driven transport have shown that the achievable confinement is sensitive to the assumed temperature just inside the edge transport barrier in H-mode plasmas (a region known as the pedestal). Recent modeling has been successful at reproducing the height of the H-mode pressure pedestal by combining theoretical predictions of stability limits with an empirical scaling for the width of the pedestal region. However, further tests are required. In addition, many other pedestal structure models are now emerging, which should be tested against experimental data from a range of devices. In particular, models are needed that dynamically capture the evolution of the pedestal when 3-D fields are applied or when transients occur. Once validated, these models should be coupled with models for heat and momentum transport in the core region to produce a comprehensive, predictive confinement model.

Other outstanding issues include the effects on the quality of the H-mode pedestal from (1) helium or hydrogen operation, (2) the near-unity ratio of input power to H-mode threshold power, (3) the relatively small separation between the primary and secondary separatrices, and (4) high opacity to edge fueling.

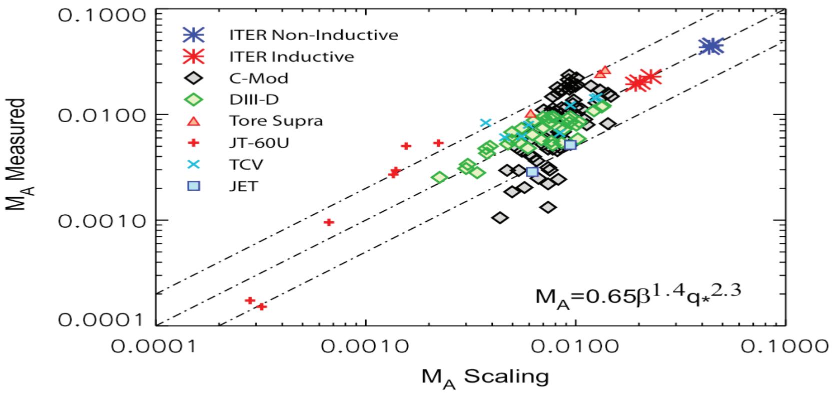


Figure 6. Multi-machine database scaling of spontaneous rotation, where M_A is the Alfvén Mach number, defined as the ratio of the plasma flow speed to the Alfvén speed. (Figure reproduced from J.E. Rice et al., Nucl. Fusion 47 [2007] 1618–1624.)

Improved understanding of the effect of non-axisymmetric fields on plasma confinement

Intrinsic and intentionally applied 3-D magnetic fields will be present in ITER, potentially having an impact on plasma confinement. Two main sources of 3-D fields present the largest uncertainties: toroidal variation of the magnetic field (known as toroidal field ripple) and non-axisymmetric fields applied for suppression of edge instabilities. A critical issue for ITER is the maximum amount of toroidal field ripple that can be tolerated while still achieving its fusion power and steady-state missions. Toroidal field ripple in ITER will arise from (1) the finite number of toroidal field coils and (2) ferromagnetic materials in the test blanket modules. Toroidal field ripple can decrease the toroidal rotation and pedestal height of H-mode plasmas. To address this issue, a model of ripple-induced losses of energetic and thermal particles and of ripple-induced toroidal rotation and/or drag must be developed. This model should be validated through detailed experiments that characterize changes in the toroidal rotation, the H-mode pedestal structure, and plasma confinement in ITER-like conditions (viz., low torque injection and low collisionality).

The application of small 3-D magnetic fields has been shown to be an effective means for suppressing edge instabilities that are generally present in H-mode plasmas. Theoretical predictions that applying such 3-D fields will induce drag and/or torque have been experimentally confirmed. Because of the importance of rotation in both the transport and the stability of burning plasmas, further experiments are needed to characterize and validate new models of these effects. For example, the counter-current offset to rotation from neoclassical toroidal viscosity could potentially cancel out externally driven rotation in the co-current direction and must be documented for extrapolation to future devices.

Develop capability to model conditions for transport barrier formation and transport sufficiently well for simulating potential ITER scenarios

Achieving the Q = 10 mission on ITER will be greatly accelerated if the capability exists to accurately simulate the full evolution of ITER plasma development, sustainment, and shutdown. Particular issues of interest are accurate prediction of thresholds for achieving or losing H-mode confinement, characterization of transport in both transient and stationary conditions, and validation and integration of the newly developed models.

H-mode Access: Access to the H-mode is essential for ITER to fulfill its experimental mission. A high-priority issue is the determination of the threshold heating power required for attaining several regimes: (1) the standard H-mode, (2) steady H-mode plasmas with "good" confinement, as determined by standard scaling relations, and (3) H-mode access during the plasma current ramp-up/down phases. These studies need to be done in ITER-like plasma conditions, and strategies should be developed for minimizing the heating power. How the required heating power for these three regimes scales with isotope mass and species (e.g., hydrogen and helium plasmas) is also of high value for the nonnuclear phase of ITER.

Research should focus on determining a physical basis for extrapolating H-mode power thresholds accurately. Experimental work would require emphasis on edge diagnostics, particularly those that measure profiles of relevant quantities such as density, temperature, and flows. Characterization of edge fluctuations leading up to L-H transitions is also important. Ample auxiliary power should be available in multiple forms (neutral beams, radiofrequency waves, etc.). Efforts should be made to resolve differences in power threshold results that may arise with different heating schemes. A major goal of the theoretical and computational efforts should be to simulate the trigger mechanisms and dynamics of an L-to-H transition and to validate results against experimental data. To facilitate comparisons across machines as well as between experiment and theory, a database containing full profile information should be assembled and made freely accessible to researchers.

Transient Conditions: To develop plasma current ramp-up and ramp-down scenarios for ITER, it is important to have a reliable prediction for the thermal electron diffusivity (which governs the electron temperature profile). Several ITPA groups have recognized this as a high-priority topic. Uncertainty concerning thermal transport during current ramp-up and ramp-down can be greatly reduced by coordinated research on this topic in available tokamaks. To be most relevant to ITER, it is desirable to use radiofrequency heating in place of neutral beam heating during the current ramp-up and ramp-down. Sawtooth "mixing" and temperature recovery between sawtooth crashes constitute another transient condition that may be of more importance in ITER than in present devices. To study transport within the sawtooth mixing region, high temporal resolution is required for measurements of magnetic pitch angle, ion temperature, and flow speed. Experiments need to vary between thermal (e.g., ohmic or electron cyclotron heating) and fast ion heating to validate models of the effect of supra-thermal sawtooth stabilization. Like sawteeth, edge localized modes (ELMs) will effectively produce significant radial transport in the periphery, and, by affecting the outer "boundary condition," the influence of ELMs can propagate into the plasma core.

Transport Barrier Formation: The fusion gain in ITER can be enhanced by the presence of internal transport barriers (ITBs) if sources of velocity shear and magnetic field shear can be produced. Possible rotation sources are discussed in the previous section. There are some indications that internal transport barriers can be generated, without velocity or magnetic shear, by pellet injection or off-axis ion cyclotron resonance heating. For future reactor-relevant regimes, it would be useful to clarify which aspects of neutral beam injection are relevant for the formation of internal transport barriers (for example, velocity shear versus energetic particles). Furthermore, experiments need to demonstrate internal transport barriers induced by magnetic field shear in stationary discharges, possibly using lower hybrid current drive or electron cyclotron current drive. Exploration is needed of electron temperature internal transport barriers and of ITB regimes with equal ion and electron temperatures. The effect of internal transport barriers on the stability limits needs to be documented. Two other critical research needs are control of the ITB radial location (to optimize the bootstrap current) and the regulation of impurities, which often accumulate following ITB formation.

Validated Models: The linking of experiment, theory, and modeling to understand core transport can be organized under the methodology of validation. Synthetic diagnostics also need to undergo verification and validation. Validation should be pursued on a hierarchy of devices over which these complexities are graduated. In regard to improving theory, a better conceptual understanding of the saturation of turbulence and the means to appropriately characterize it is clearly needed. Finally, bringing all of this together into a validated, integrated model of transport in ITER is necessary.

CREATING A SELF-HEATED PLASMA

Producing plasma conditions conducive to sufficient alpha particle generation for self-heating in ITER poses a myriad of challenges. To achieve such conditions, careful control of a variety of parameters (e.g., fuel and impurity content, magnitude and timing of externally applied heating) is required. In present-day devices, trial-and-error approaches have established methods for controlling critical aspects of operation to achieve scenarios that have characteristics similar to those needed for Q = 10 operation in ITER. However, the complexity of the ITER device will limit the ability to develop self-heated plasmas by trial and error. Instead, pre-experiment simulations will be relied upon to develop, test, and tune operational scenarios prior to use in ITER. In this regard, research is still needed to improve the physics understanding (and hence simulation capability) of the processes important in the evolution of the plasma state. In addition, further research is needed to improve the physics basis for recently developed scenarios that offer the potential of extended duration or higher-Q operation in ITER.

CURRENT STATUS

Issues

Tokamaks such as ITER use the toroidal current of the plasma to generate the confining magnetic field. Projections indicate that to achieve conditions conducive for burning plasma operation, a plasma current of ~ 15 MA will be required in ITER. Achieving Q = 10 operation in ITER therefore requires several steps: an initial plasma must be created via ionization of injected gas; closed magnetic confinement surfaces (known as flux surfaces) must be formed; the plasma current must be increased (current ramp-up) through either transformer action (ohmic current drive) or other external means; and heating must be applied to achieve thermonuclear temperatures. Also, protecting the machine integrity requires a reliable means for slowly shutting down the plasma, allowing the thermal and magnetic energy to be released on a time scale compatible with protection of the plasma facing components and other structures. The various stages of an ITER Q = 10 plasma are illustrated in the simulation shown in Figure 7.

Although present-day devices have made significant progress in developing individual techniques and integrated scenarios for achieving plasma conditions consistent with Q = 10 operation in ITER, research is still required on the specific application of these techniques and scenarios in ITER. Research issues in this area can be categorized in the following areas:

- Performance capabilities of ITER in non-DT phases.
- Techniques for plasma breakdown and current ramp-up/down consistent with operating constraints on ITER.
- Capabilities for heating, fueling, and power exhaust in burning plasma conditions.
- Accessibility of enhanced performance regimes in ITER.

For ITER to achieve its Q = 10 goal, there are uncertainties and issues that need to be resolved, some of which are described in this section. However, the consensus among the world fusion research community is that none of the issues are insurmountable or show stopping and therefore ITER has a very good probability of achieving this major objective.

Accomplishments

Most of the elements required for formation and control of a high-performance ITER plasma have been and are routinely achieved in the international tokamak research program. For example, many techniques have been developed to prepare the condition of the discharge vessel, allowing the formation of a high-purity hydrogenic plasma. Techniques for controlling plasma shape, internal inductance and, to a lesser degree, current profiles during ramp-up, flattop, and ramp-down are routinely employed throughout the worldwide tokamak program. Their success mirrors the depth of understanding of the MHD behavior of high-temperature plasmas. The scientific and technological basis for heating plasmas to thermonuclear temperatures and pressures with the use of neutral beams and radiofrequency wave heating at electron and ion cyclotron frequencies is well developed. Neutral beams have been used to heat plasmas to over 40 keV in TFTR, well over the ~ 25 keV central temperature required for ITER, while heating at ion cyclotron frequency has been used in Alcator C-Mod to increase the plasma pressure to 1.8 atmospheres, essentially equaling ITER's operating pressure. Techniques for gas fueling and fueling by injection of frozen hydrogenic pellets have been developed and, together with active pumping, are used to control the plasma density. An important result is the discovery that fueling from the high-field region of the vacuum vessel can be used to improve the effective fueling depth and offer some degree of control over the density profile. Excellent progress has also been made in the control of plasma heat exhaust, one of the most daunting issues to be confronted as plasmas move into the long pulse, burning regime. A particularly important development was the observation and understanding of "divertor detachment," in which a large fraction of heat flux flowing into the divertor region is dissipated before contacting the divertor target plates. A basis for extended pulse operation has been established with the discovery in the last few years of the so-called "hybrid mode," a regime of improved confinement in which the central safety factor $q(0)$ is clamped near unity. By increasing $q(0)$ above unity, regimes have been realized in which a large fraction (>50%) of the current is self-driven (the so-called bootstrap current). These are often augmented by the formation of "transport barriers," regions of the plasma that are characterized by relatively steep gradients and low transport. In combination with current drive from neutral beams and/or radiofrequency

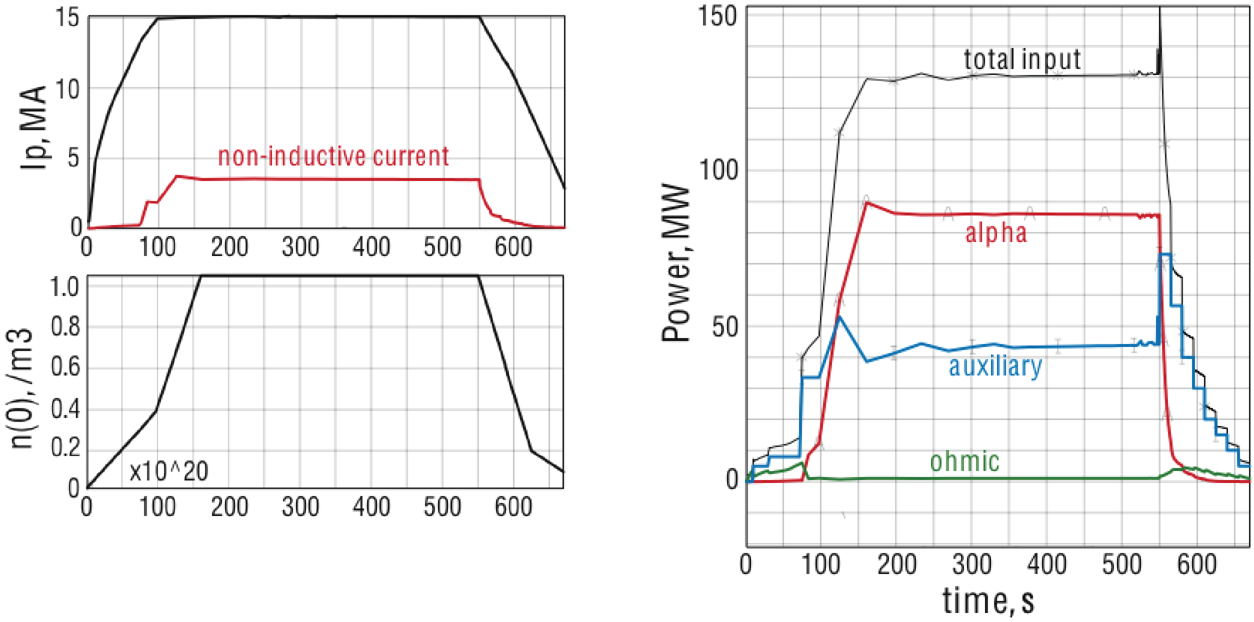


Figure 7. Simulation of an ITER discharge. The current ramp-up, flattop, and ramp-down phases are illustrated in the upper left panel. Fueling is added during the ramp-up and flattop phase to bring the density up to slightly over 1×10^{20} m^{-3} during the flattop phase. The alpha power trace in the right panel indicates that the total fusion power is 425 MW. Careful programming of the auxiliary power is required to produce a stable burn and a "soft landing" at the end of the discharge. (Figure courtesy of C. Kessel.)

waves, the bootstrap current points the way toward steady-state non-inductive tokamak operation. From this viewpoint, the theoretical prediction and subsequent experimental verification of the bootstrap current is considered to be one of the most significant and important results from the international tokamak research effort.

SCIENCE CHALLENGES, OPPORTUNITIES, AND RESEARCH NEEDS

Evaluation of ITER performance capabilities in non-DT phases

An extended period of operation in hydrogen and helium, followed by operation in deuterium, is planned before full D-T operation in ITER. The pace at which Q = 10 operation can be achieved in the D-T phase will be strongly influenced by the research that can be done during the preceding non-DT phases. For example, achieving robust H-mode operation during the non-DT phases would enable the development of techniques to handle critical issues (such as ELMs and stabilization of neoclassical tearing modes) early in the research program.

Each fuel presents special issues relative to the baseline D-T operation for which ITER was designed. For example, access to the reference ELMy H-mode is generally more difficult to obtain in hydrogen than in heavier fuels such as deuterium or a mixture of deuterium and tritium. Similarly, ion cyclotron resonance heating scenarios that are based on heating a minority helium-3 component and/or tritium at the second harmonic of the cyclotron frequency must be adapted to the conditions of the initial operating phase. Research on the requirements for developing H-modes for the early operating phase of ITER is required, preferably resulting in a validated theoretical model for the L-mode to H-mode transition. This is particularly important since present empirical estimates indicate that it will be difficult to achieve H-mode in the non-DT phases in ITER. In addition, development of appropriate ion cyclotron resonance heating scenarios for the early phases of the ITER research plan is needed to ensure that these phases will be as relevant as possible to subsequent D-T operation.

Techniques for plasma breakdown and current ramp-up and ramp-down consistent with ITER operating constraints

Vessel Cleaning at Full Field: A relatively prosaic but very important issue for ITER is developing the means to clean the vessel walls of loosely bound elements such as carbon, beryllium, and water vapor prior to tokamak operation. During the breakdown and startup phase of fusion devices, lightly bound elements are released from the wall, which may cause the discharge to collapse radiatively before reaching temperatures high enough to permit the induced electric field to drive significant current. This problem is well known in existing tokamaks, and various methods of in-situ cleaning of the chamber walls have been developed. However, most of these methods are not applicable to ITER because they have been developed for pulsed machines in which the toroidal field is typically not present during the cleaning process. In ITER, superconducting magnets produce the 5.3 tesla toroidal field, and it is highly desirable to maintain the field constant at its design value. Thus, the research issue here is to develop an effective means to clean the ITER vessel walls while maintaining a constant 5.3 tesla toroidal field.

Ramp-up and Ramp-down: Present ITER scenarios call for the plasma current to be ramped up in 65-100 seconds and a diverted plasma to be formed relatively early during the ramp-up. Heating power is to be applied at or soon after the plasma becomes diverted to keep the internal inductance low and avoid vertical instability. The transition to H-mode is to be avoided until late in the ramp-up. A key question is then how to manage this ramp-up scenario to minimize the transformer consumption (i.e., resistive volt-seconds), so that the maximum volt-seconds will be available at full plasma current. To answer this question with more extensive modeling, one needs to know the plasma transport properties — e.g., energy, particle, and impurity diffusion coefficients — during this transient ramp-up phase. The problem is complicated by the fact that the total current radial profile is evolving on the ramp-up time scale, and this will feed back on the transport, which in turn affects the current profile. The effectiveness of the heating and current drive sources for reducing the transformer consumption during ramp-up needs verification with respect to the associated transport and resistive MHD limitations.

The current ramp-down in ITER is complicated by the requirements to maintain MHD stability (i.e., achieving a soft landing) and avoid additional flux consumption while exiting from burn. Here again, knowledge of the particle and energy transport coefficients is necessary for modeling this phase of the discharge. Additional issues include whether and when the plasma makes a transition back to L-mode, how to keep the plasma coupled to the mid-plane antennas for heating and to the divertor strike points for particle and power handling, and how to keep the plasma diverted. While "normal" ramp-down is described here, these issues also arise for "emergency" shutdown if an incipient disruption cannot be mitigated and the discharge must be terminated. Thus a variety of initial conditions for the ramp-down phase must be considered.

Heating, fueling, and power exhaust in burning plasma conditions

Auxiliary Heating: The auxiliary heating and current drive schemes planned for initial operation of ITER feature 20 MW of electron cyclotron power at 170 GHz, 20 MW of ion cyclotron power at 50 MHz and 33 MW of neutral beam injection at 1 MeV energy. Issues related to electron cyclotron resonance heating and neutral beam injection are largely technical. Although a 1 MW prototype gyrotron operating at 170 GHz has been successfully tested for long pulse, the technology is still in its infancy, and the reliability of 24 such tubes operating simultaneously to deliver the required 20 MW of electron cyclotron power needs confirmation. Although our ITER partners are adequately addressing gyrotron technology for ITER, the absence of a US effort on gyrotrons for ITER raises a concern about the basis for the US to proceed toward a DEMO, which in all probability will require electron cyclotron resonance heating technology. In the case of neutral beam injection, many years of effort have failed to reach 1 MeV beams at the requisite current density. There are plans to build a full test stand in Europe, and an aggressive attack on this issue is underway. Neutral beam injection has been the most reliable and widely used heating and current drive source in the international tokamak effort and was used to create record-high ion temperatures in TFTR (see Figure 8). Consequently, extrapolations of the tokamak database in critical areas such as torque generation, rotational control of stability, fast ion physics, and active beam diagnostics are considered more credible than for other heating and current drive methods.

Ion cyclotron resonance heating is also effective in heating tokamak plasmas, as shown in the example from Alcator C-Mod (see Figure 9), where it has been used to produce record plasma pressures. However, unlike electron cyclotron resonance heating and neutral beam injection, where the technology is still in development but the heating physics is relatively straightforward, ion cyclotron resonance heating is based on relatively secure technology with some uncertainty in the physics. The main issue is that ion cyclotron antennas must be in contact with the edge or scrape-off layer (SOL) plasma so that the antenna impedance can be matched at antenna voltages below breakdown levels, typically 50 kV. Consequently, ion cyclotron resonance heating antennas form radiofrequency sheaths in the scrape-off layer that can accelerate ions into the plasma facing materials at energies of a few 100 eV, well in excess of sputtering thresholds. Sputtered impurities

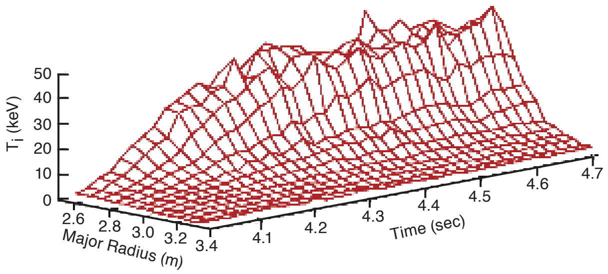


Figure 8. Creation of record high ion temperatures (> 40 keV) in TFTR by neutral beam heating. (Figure courtesy of E. Synakowski and R.E. Bell; for a related article, see D. Mansfield et al., Phys. Plasmas 3 [1996] 1892.)

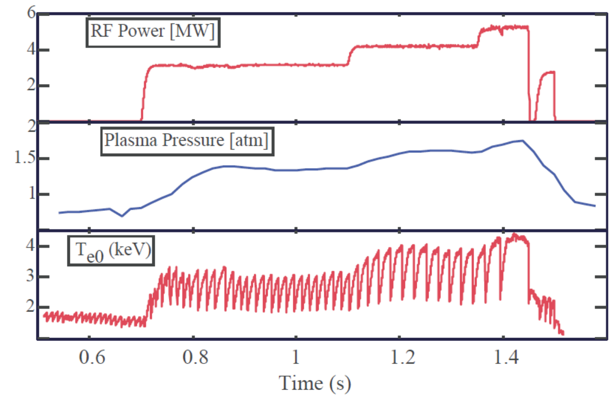


Figure 9. RF power at the ion cyclotron frequency in an Alcator C-Mod plasma discharge increased the plasma pressure to 1.8 atmospheres, the same pressure at which ITER will operate. (Figure reproduced from S. Scott et al., Nucl. Fusion 47 [2007] S598.)

can enter the plasma core and diminish its performance either through radiation, in the case of heavy first-wall materials such as tungsten, or through fuel dilution for light impurities such as beryllium or carbon. Progress is being made both in understanding the physics of radiofrequency sheath formation and in designing ion cyclotron resonance heating antennas that can reduce sheath formation. However, given the importance of ion cyclotron resonance heating to the success of ITER (since it is the only method that heats ions), the research effort needs to be intensified with dedicated experiments and validated simulations of the antenna-plasma interaction, including the physics of radiofrequency sheath formation.

Fueling: Fueling efficiency and density profile control require that ITER be fueled well beyond the separatrix. The means for doing this is pellet injection, where, for maximum efficiency, pellets are launched from the inner (high field) region of the ITER vessel. Predictive understanding of the density profiles achieved with this arrangement requires quantitative knowledge of pellet ablation and penetration physics, in the presence of energetic ions, as well as of particle transport. The issue is complicated by the burning plasma environment of ITER, which couples the density produced by pellets to the alpha particle production rate. The effect of pellets on ELMs (possibly beneficial if "pellet pacing" is effective in reducing the size of ELMs) and other MHD activity also requires further research to make reliable predictions of the efficacy of pellet fueling in ITER.

Fueling by pellets offers the further advantage of reducing the tritium inventory by injecting tritium-rich pellets, while fueling with gas at the edge with deuterium (and impurities) to maintain optimal divertor conditions. It then should be possible to provide a degree of control over the burn in ITER by feedback control of the isotopic mix. The time scale over which the mix can be varied is an important consideration and this requires additional research in existing pellet-fueled devices.

The overall particle control issue, which includes fueling, is critical for the sustainment of burning plasmas. Pumping, the introduction of radiating impurities, the introduction of unintentional impurities, the removal of helium, the fuel mixture and core fueling, and the radiating divertor and overall power handling all must be simultaneously achieved for the burn to be sustained.

Heat Exhaust: Managing the heat load, both during the nominal steady state and transients, represents one of the biggest challenges to ITER. The largest heat load is on the divertor targets and, at 5-10 MW/m^2, is at the limit of what is technologically feasible, especially considering the difficult plasma environment. Transient events such as ELMs and loss of detachment can severely limit the target lifetime. To extend the lifetime, research is needed on critical heat removal issues, including how to control the power flowing to the divertor through feedback. An additional issue is the viability of the plasma facing components after large-scale surface damage and loss of conforming surface geometry (relative to the magnetic field) due to repetitive ablation/melt loss from ELMs. The physics of radiative divertor solutions — including effects of opacity, impurity radiation (and effect on the core), and momentum transfer — also needs elaboration to develop confidence in the projected performance of the divertor design for ITER.

Tritium Retention: Retention of tritium in first-wall components had been identified as a critical issue for ITER since the earliest days of the ITER design. The problem is especially acute for carbon plasma facing components due to co-deposition, the tendency of carbon to form difficult-to-remove hydrocarbon deposits in the vessel chamber. The main alternative material for the high-heat-flux components is tungsten, and it is important to quantify hydrogen retention rates in tungsten, as well as in beryllium, which is the material for the first-wall components, excluding the baffles and divertor. While laboratory studies are useful for this task, it also will be necessary to confirm retention rates under ITER-level plasma heat and particle fluxes.

Developing the means for removing tritium from the plasma facing components is also a critical issue. Proposed methods include thermal oxidation of hydrocarbon films and thermal recovery from beryllium. The feasibility of intense, localized heating of plasma facing components should be assessed, as well as "operational" recovery strategies involving the use of strong divertor pumping and moderate-level disruption flash heating.

Dust production is another important issue related to plasma facing components, particularly for safety. Methods of measuring the production rate and characterizing the composition of dust in operating tokamaks need to be developed, as well as mitigation and removal methods.

Accessibility of enhanced performance regimes in ITER

The ITER research plan envisions two phases. The primary goal of the first phase is to achieve a burning plasma, generally defined as one in which heat liberated from fusion reactions and carried by the fusion-product alpha particles is at least twice that produced by auxiliary sources; this corresponds to an energy gain Q ≥ 10. In its first phase of operation, ITER plans to achieve Q=10, sustained for a 300-500 s duration. In its second phase, the objective of ITER is to aim at achieving a steady-state burn, specifically, to achieve a gain of Q ≥ 5 with fully noninductively driven current. Achieving this goal is foreseen to occur in two main stages of increasing duration: (1) "hybrid" operation with a residual inductive fraction of the total current, but sufficient inductive volt-seconds for ~ 1000 s pulse, and (2) a steady-state mode, where there is no residual inductively driven current and the plasma configuration can in principle be sustained indefinitely (although practical limitations enforced by ancillary equipment such as cooling towers might limit the duration to ~ 3000 s.) In contrast to the goal of achieving Q ~ 10 for relatively short pulse, the physics basis for achieving these two extended regimes of operation is less well developed and therefore rich in opportunities for contributions by the US fusion research community.

Hybrid Regime: The hybrid regime is characterized by a broad, flat current profile with a central safety factor of $q(0) \geq 1$. Due to the broad region of reduced magnetic shear, global confinement can exceed typical H-mode levels. In addition, the broader current profile tends to result in higher stability limits. Hybrid plasmas are characterized by a non-negligible residual contribution from the inductive current and are therefore not candidates for a steady-state mode of operation in ITER. However, since they have better than normal H-mode confinement as well as a higher stability limit, enhanced performance can be achieved at currents lower than those for the reference high-Q mode of operation. The savings in volt-seconds can then be applied to extend the pulse length. The hybrid regime is thus seen as an intermediate step between the high-Q, 300-400 s pulsed operation and a moderate-Q, fully steady-state mode of operation. Successful development of the hybrid regime in ITER would facilitate the initial stages of nuclear testing. A key issue in realizing the hybrid regime on ITER is whether the broad current profiles systemic to this regime can be achieved in ITER. Analysis of experimental results suggests that the sources of current drive alone cannot account for the observed broad current profile. Similarly, simulations for ITER indicate that the day-one mix of heating and current drive capability is insufficient to robustly maintain the requisite flat shear for an extended duration. The mechanism leading to the low shear with $q(0) \geq 1$ in present-day devices is correlated with the appearance of benign MHD instabilities, for example, the appearance of fishbone activity in ASDEX-Upgrade and a low-level 3/2 neoclassical tearing mode in DIII-D. These instabilities prevent the central safety factor from dropping below unity, which results in stabilization of the sawtooth instability normally found in H-mode discharges. More research is required to determine if this "natural" current profile broadening can be expected in ITER.

Experimental results also suggest that performance in the hybrid regime is adversely affected by operating characteristics expected to be encountered in ITER, such as low rotation and T_e/T_i =1. Improvements in the ability to simulate these changes using theory-based transport models is needed to confirm whether improved performance can be realized in ITER.

Steady-state Operation: Steady-state operation of ITER is highly desirable, not only to carry out the nuclear testing part of the ITER mission, but also to demonstrate the tokamak's potential to form the core of a practical fusion reactor. In general, steady-state operation of a tokamak is based on replacing the current normally generated by induction with current generated by other means. Although toroidal current can and will be produced in ITER by neutral beam injection and the various forms of radiofrequency current drive, the efficiency of these systems is too low to generate a substantial fraction of the total current, i.e., the power required to replace the normal inductive current would be comparable to the total power derived from the fusion reactions. Thus, the approach to steady state for ITER, and indeed for a tokamak reactor, is to maximally exploit the "bootstrap current," which is naturally produced in a toroidal plasma by the plasma's temperature and density gradients. Achieving this "self-organized" state requires operation at relatively high beta, which means that steady-state tokamak operating regimes represent a delicate balance between generating sufficient bootstrap current and respecting MHD stability limits.

Good progress in achieving practical, near-steady-state tokamak operating regimes has been made in the worldwide tokamak research effort, in particular, in the DIII-D tokamak in the US. As in the case of hybrid modes, the magnetic shear appears to play a key role in forming high-bootstrap-fraction discharges. In DIII-D, fully noninductive discharges have been obtained with a bootstrap fraction of ~ 60% in regimes with flat shear and $q_{\min}$ ~ 1.5. Another example of near-steady-state operation in low-shear regimes is provided by the so-called high poloidal beta regimes obtained in JT-60U. In these discharges, the central safety factor remains in the range $1 < q(0) < 2$ and a bootstrap fraction of ~ 50% is generated, consistent with achieving a high poloidal beta value. A weak transport barrier forms in both DIII-D and JT-60U low-shear discharges and is instrumental in generating a high fraction of bootstrap current.

A second basic mode relevant to steady-state operation is associated with strong shear reversal. This mode has been explored in a variety of devices including JET, JT-60U and DIII-D. In this mode, the central safety factor increases above 2 and, in the case of strong shear reversal, can rise well above 2; in the extreme "current hole" regimes of JET and JT-60U, the value of $q(0)$ can rise above 20. Such strong shear reversal regimes are favorable for formation of internal transport barriers as manifested by the appearance of steep gradients in the temperature and/or density profiles. Internal transport barriers in conjunction with low central current density (and therefore low poloidal field) are conducive to high bootstrap current, which is desirable for efficient steady-state operation. However, the beta limits for these regimes tend to be lower than for the relatively flat shear regimes due to the large pressure gradients, and MHD stability becomes problematic. In addition, the increased energy confinement and particle confinement may result in impurity accumulation, which would lead to an unacceptable degradation of performance in ITER or a reactor. And, finally, such regimes have poor central fast ion confinement and are vulnerable to fast ion instabilities. Thus, the applicability of regimes with strongly reversed magnetic shear to achieve high-performance steady-state operation in ITER cannot be assessed without additional research and optimization.

Simulations indicate that achieving either flat or reversed-shear steady-state regimes in ITER will be difficult because the day-one heating and current drive mix provides little or no ability to modify the q-profile, yielding a mostly monotonic increasing profile due to the deposition locations for neutral beam current drive and electron cyclotron current drive. This is especially the case for neutral beam current drive since the 1 MeV neutral beams tend to produce central current drive, which will prevent $q(0)$ from rising above unity. Tilting of the beam lines improves the capability of neutral beam current drive to drive current off axis, but current profiles calculated for the latest design are still confined to ~ 30% of the minor radius. Experiments on these regimes in ITER may be possible during the early current ramp phase when reversed-shear configurations could be formed, but these would likely be transient without an assured off-axis method of current drive.

Under the assumption that an appropriate off-axis current drive system could be deployed on ITER, an intensive research effort would be needed to prepare the basis for achieving hybrid and steady-state regimes in ITER. The main requirements for carrying out this research are the availability of a high performance, well-diagnosed tokamak with pulse length much in excess of the resistive current relaxation time, adequate heating to ensure that beta limits can be challenged, and sufficient off-axis current drive power to ensure that steady-state (on the resistive time scale) scenarios with minimum q significantly greater than unity can be maintained. It is not clear that any device in the current worldwide portfolio of tokamaks is well matched to these requirements. In view of the importance of developing suitable scenarios for steady-state tokamak operation, not only for ITER but also for the step beyond, this question must be addressed if no existing device can adequately fulfill the requirements.

CONTROLLING AND SUSTAINING A BURNING PLASMA

Control is an important element for ensuring efficient and robust operation of large, complex systems. This is particularly true in a burning plasma device such as ITER, which will require extensive control solutions, some yet to be developed. These control solutions must simultaneously sustain core plasma conditions that are conducive to fusion power generation, provide methods for exhausting the produced energy without damage to internal materials and components, and ensure safe operation of all main tokamak and auxiliary systems (see Figure 10). Control research needs for ITER are discussed in the present section under Theme 1. Control research focusing on the needs of DEMO and an eventual power reactor is discussed in the Theme 2 chapter of this Report.

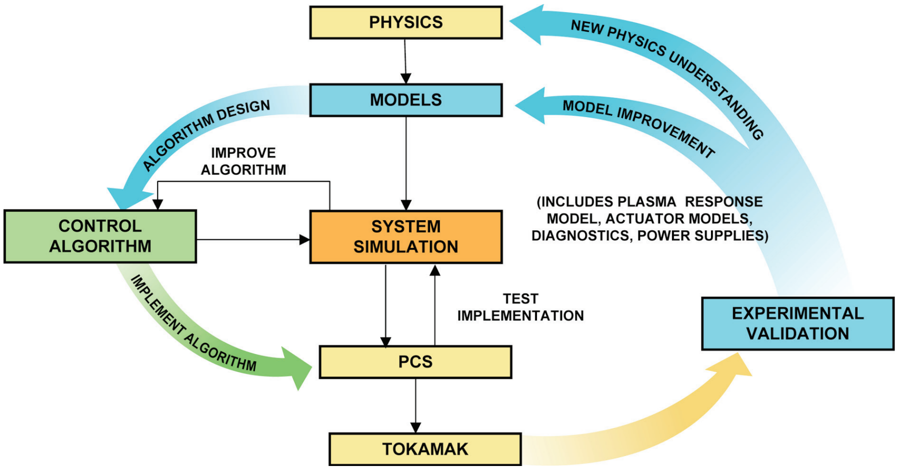


Figure 10. Control involves the elements for implementing regulation of a fusion device, active solutions to accomplish control scenarios and maintain performance, feed-forward and feedback algorithms, and a Plasma Control System (PCS) for supervisory control. (Figure courtesy of E. Synakowski; for a related article, see D. Mansfield et al., Phys. Plasmas 3 [1996] 1892.)

CURRENT STATUS

Issues

ITER will require plasma control solutions with higher levels of performance and reliability than are achieved in present-day devices. However, due to the size and complexity of the ITER device, both the ability to dynamically measure and control specific attributes of the ITER plasma will be limited. In addition, its nuclear mission imposes requirements on performance certification far beyond those of any device yet constructed. Although present research programs and presently operating devices have already produced general solutions for some of the control challenges expected in ITER, many remain to be solved, for which substantial effort will be required.

Control issues for ITER can be categorized as:

- Providing robust, reliable plasma control in burning plasma conditions.
- Handling off-normal events and faults in a safe, reliable manner.
- Developing model-based control algorithms.

These issues share several common requirements. Because ITER must qualify its control performance well before operational implementation, all control algorithms must be model-based, requiring development of *validated control-level models* for design of relevant systems in active control or detection/response loops. Sufficiently *detailed simulations* and associated models must be developed to verify both performance and implementation of controllers prior to use. *Control algorithm design* approaches and solutions must be created to satisfy the demanding performance requirements of ITER. In many cases, this will require the development of new mathematics and algorithmic understanding at the cutting edge of control science. *Real-time computational solutions* are required for analysis and interpretation of plasma and plant state (e.g., real-time analysis and prediction of proximity to stability boundaries) as well as implementation of complex control algorithms. Innovative control *actuators and diagnostics* will likely need to be developed with effectiveness, dynamic performance, and accuracy. *Experimental demonstrations* of control schemes and specific control algorithms are essential to fully confirm the solution before application to ITER. *Improved physics understanding* may still be required, including sufficiently detailed computational physics models and control-level models. Control *models* frequently require far less accuracy and/or precision than that for detailed physics codes (although measurement accuracy and precision requirements tend to be very high for real-time control). *Improved mathematics and algorithmic understanding* are required in many cases to develop the required control schemes and controller designs. The engagement of cross-disciplinary expertise, including physics, control mathematics, and fusion system engineering, will be essential in filling these research gaps.

Providing robust, reliable control of plasma

Plasma Startup and Shutdown: Achievement of robust, non-disruptive startup and shutdown scenarios for ITER represents a challenging operational control problem owing to the tight constraints on the available transformer action (i.e., volt-seconds), superconducting coil voltages and currents, plasma shape, and allowable disruptivity. Development and experimental demonstration of startup and normal shutdown solutions will be required for each ITER operating scenario. Rapid shutdown solutions must also be developed for an unscheduled demand shutdown or to preempt an impending off-normal event.

Operating Regime Regulation: Regulation of the ITER operating regime includes the equilibrium shape and position state, bulk quantities (such as plasma current and stored energy), various profiles (including current density, pressure, density, and rotation), and the divertor configuration. Because ITER scientists recognize the importance of using robust model-based multivariable control solutions for shape and stability control, this approach must be demonstrated and qualified in operating devices prior to ITER operation, along with the pulse verification simulations envisioned to precede each ITER discharge. Solutions for equilibrium control consistent with the ITER operating environment must also be developed and demonstrated.

Kinetic Control: Achieving self-consistent solutions that combine adequate D-T fusion power generation in the core and sufficient heat dispersal for protection of the plasma facing materials requires methods for controlling the kinetic state of the plasma, including fueling and divertor regulation. Fueling solutions consistent with ITER requirements (e.g., high-field-side deeply injected pellets, ~1.0 cm^3 solid D-T) must be developed, although full demonstration will likely await the ITER burning plasma. Methods for active control of divertor-target heat flux, radiation state, and degree of detachment all must be developed and demonstrated. Although ITER will probably operate in the thermally stable regime, methods for demonstrating burn regulation (e.g., fueling regulation coupled with saturated MHD amplitude regulation to modify transport) will be required to ensure minimal excursions arising from operating point fluctuations.

Stability Control: ITER operation in high-performance regimes requires active and reliable stabilization of a number of instabilities, including the resistive wall mode, neoclassical tearing modes, axisymmetric instabilities, ELMs, and possibly also sawteeth and energetic particle-driven modes. Development of active stabilization solutions for these and other instabilities will require greatly increased understanding of all relevant modes and their stabilization mechanisms, along with experimental demonstrations in multivariable, highly coupled, integrated forms prior to operation of ITER. Solutions for controlling ELMs that are consistent with ITER constraints are particularly critical, owing to predictions that unmitigated Type I ELMs will ablate the divertor targets at an unacceptable rate. Control solutions for specific axisymmetric instabilities, resistive wall modes, and neoclassical tearing modes must be developed and qualified with sufficient reliability and performance to produce the required low level of disruptivity. Axisymmetric control solutions may also be required for the regulation of runaway electrons following a disruption.

Fusion Plant: Certain limited elements of a power plant fuel and breeding cycle will be tested in ITER, including Test Blanket Modules (TBM) for breeding tritium. Control requirements associated with these elements include development of TBM operation control solutions and remote maintenance solutions, possibly including accelerated access and replacement of major components such as divertor cassettes.

Handling off-normal events and faults in a safe, reliable manner

The success of ITER will critically depend on the control system's ability to minimize the frequency of off-normal events and to respond effectively to minimize damaging effects. A key part of this solution is to demonstrate reliable control in the proximity of stability boundaries, including excursions expected during nominal operation, as well as off-normal or fault-triggered excursions. Reliable operation under nominally disturbed regimes (e.g., intermittent sawteeth, ELMs, or transient magnetic island growth) falls under the research category in Chapter 2. However, novel algorithmic control and response solutions beyond nominal control are also required for excursions requiring transient changes of the operating regime (e.g., response to loss of a diagnostic channel or key actuator), rapid but controlled shutdown (e.g., identification of an impending unrecoverable state), or emergency uncontrolled shutdown (e.g., identification of immediate unrecoverable state and insufficient time for controlled shutdown, often envisioned as serving to mitigate potential system damage). Enabling elements of these solutions include a quantifiably reliable systematic approach and corresponding integrated system for response to off-normal events and fault-triggered excursions, real-time predictors for proximity to key operational limits, algorithms and mechanisms for mitigating damage, and recovery/cleanup strategies to rapidly restore plant availability. Approaches must be developed to quantify performance and risk/probabilities for licensing and certification.

Developing model-based control algorithms

Design of model-based control algorithms requires both new computational tools as well as new algorithmic approaches. Computational tools are needed to produce control-level models, integrated and sufficiently comprehensive simulations, and real-time predictive models for on-line controller adjustment or operating regime identification. ITER will specifically require an integrated, axisymmetric, resistive MHD simulation code for development and testing of controllers, as well as pulse verification. Verifying pulse consistency and performance is an operational requirement of the present ITER control system specification. The models and simulations used for these purposes must be thoroughly validated against data from operating devices.

These model-based control tools will require the development of new algorithmic approaches. This research includes mathematical solutions for nominal high-performance control, as well as supervisory and off-normal response algorithms with provable reliability. This will include the development and experimental qualification of control schemes and detailed real-time implementation for nominal scenario regulation, stability control, and supervisory action with effective and provable responses to off-normal events. The reference ITER approach to integrated plasma control (validated models, model-based design, verification of performance) must be demonstrated end-to-end on operating devices.

MITIGATING TRANSIENT EVENTS IN A SELF-HEATED PLASMA

Transient events that release plasma energy in a short period of time have the potential for causing serious damage to plasma facing components and other structures in ITER. Preventing these events to the extent possible and mitigating those that occur are critical in protecting the investment in ITER as well as expediting the research program. The FESAC report entitled *Priorities, Gaps and Opportunities: Towards A Long-Range Strategic Plan For Magnetic Fusion Energy* specifically identified the control and mitigation of such transient events as critical issues for both ITER and future fusion devices.

CURRENT STATUS

Issues

Avoiding or mitigating transient plasma events in ITER is very challenging, both in the high-current Q=10 scenario, as well as in the steady-state operating regime. In general, the preferred approach is to avoid the initiation of a transient event that could result in unacceptable stresses or damage to machine components. In some cases, either due to a control system failure or inability to forecast the occurrence of a transient event, it will be necessary to take actions to reduce or mitigate the consequences of a transient event to avoid exceeding established limits on machine components. For example, mitigation techniques may result in terminating the plasma without exceeding allowable stresses on the machine.

Pursuant to the FESAC panel report cited above, the following transient events for ITER were considered:

- Disruptive termination of the plasma discharge, including the generation of runaway electrons.
- Large edge instabilities (edge localized modes — ELMs) that are associated with H-mode operation.
- Plasma instabilities that eject large bursts of alpha particles.

All of these are important for ITER, as discussed in the *ITER Physics Basis* document, as well as for DEMO, which will be discussed in the Theme 2 chapter.

Recent accomplishments and progress

Substantial progress has been made in characterizing, predicting, avoiding, and mitigating off-normal and transient events in existing high-performance plasma experiments. Highlights include:

- Suppression of disruptions with the massive gas injection technique.

- Identification and stabilization of neoclassical tearing modes by electron cyclotron current drive.
- Identification and stabilization of resistive wall mode instabilities by the application of non-axisymmetric magnetic fields.
- Suppression of ELMs by the application of non-axisymmetric magnetic fields.
- Identification of various types of ELMs and the discovery of tokamak operational regimes that are ELM-free.
- Observation of alpha particle loss due to instabilities in D-T plasmas.

SCIENCE CHALLENGES, OPPORTUNITIES, AND RESEARCH NEEDS

Disruptions

Disruptions and disruption-like loss-of-vertical-stability events, called vertical displacement events (VDEs), pose significant design and operational challenges for ITER and for subsequent reactor tokamaks in general. In ITER, the consequences of disruptions are anticipated to be more severe than in present-day devices. Electromagnetic loadings and area/size-normalized body forces increase modestly (~3 times) from present tokamaks to ITER and can be accommodated in the corresponding structural designs. However, the time-weighted energy deposition onto plasma facing components in ITER and DEMO will be an order of magnitude greater than in present tokamaks and could approach or exceed thresholds for the onset of tungsten surface melting (or carbon vaporization). Avalanche multiplication of a highly energetic electron population (known as runaway electrons) may occur during disruption, vertical displacement events, or fast-shutdown current decays (the latter effected by gas or pellet injection). The avalanche gain for this process, $G_{aval} \cong \exp[2.5I(\mathrm{MA})]$, is sufficiently high that even very minute levels of “seed” runaway current can convert to a very large population (> 5 MA) of ~10 MeV electrons. Subsequent deposition of these electrons on plasma facing components could cause considerable damage.

Relative to present-day devices, disruption avoidance in ITER is significantly more challenging (see Figure 11). Estimates indicate that large improvements in two metrics for “disruptivity” relative to present experiments are needed: viz., a 10-fold reduction in per-pulse disruption rate (discharge setup reliability) and a 1000-fold reduction in per-second disruption rate (flattop/burn sustenance reliability).

For future burning plasmas, accurate prediction of an impending disruption is a key element of both avoidance and mitigation of disruptions. It will be essential (1) to develop plasma operation and control procedures that *avoid*, wherever possible, occurrence of disruption onset, (2) to identify means to reliably *predict* pending onset of disruption, and (3) to have means available to *mitigate* the consequences of disruptions that cannot otherwise be avoided. In developing strategies for disruption prediction, avoidance, and mitigation, it will also be essential to develop quantitative metrics to *characterize and model* disruptions and disruption consequences. In all four categories, there are substantial “gaps” or open issues.

Disruption avoidance

Most disruptions are, in principle, avoidable. Many causes of disruptions are well understood, including loss of axisymmetric stability (VDE), radiative collapse (density limit), and ideal and resistive MHD instabilities (beta limit, current limit, locked modes). Their onset can be predicted on an empirical or a theoretical basis. By avoiding such conditions, present tokamaks routinely operate at pulse durations of tens of seconds without disruptions. Avoidance of disruptions in ITER will require reliable maintenance of the desired plasma configuration, detection of stability limits, corrective action when approaching stability limits, and active suppression of instabilities. The associated research needs include:

- Diagnostics and actuators for control of the pressure and current profile, particularly in conditions in which these profiles are largely determined by internal processes (e.g., self heating, large bootstrap current).
- Diagnostics to detect stability limits (see disruption prediction in Figure 11).
- Actuators for direct control of instabilities, including localized current drive, localized rotation drive, and non-axisymmetric magnetic coils.
- "Intelligent" control algorithms that can stably maintain the plasma state away from stability limits and recover from changes in the plasma state caused by non-disruptive instabilities and other unplanned occurrences. If an instability and subsequent disruption cannot be avoided, the control system must be able to attempt a controlled "soft shutdown" of the discharge.

Much of the development of diagnostics, actuators, and control can be carried out in existing short-pulse facilities. It will be important for the new generation of superconducting tokamaks (EAST, KSTAR, SST-1, and JT-60SA) to demonstrate that "disruption-free" long-pulse or steady-state discharges with high performance, as needed for ITER and DEMO, can be realized. However, a full test of plasma control and disruption avoidance in a self-heated plasma situation will only become possible in ITER. Disruption avoidance with high reliability must be developed and demonstrated in ITER before proceeding to DEMO, where the requirements are even greater than for ITER.

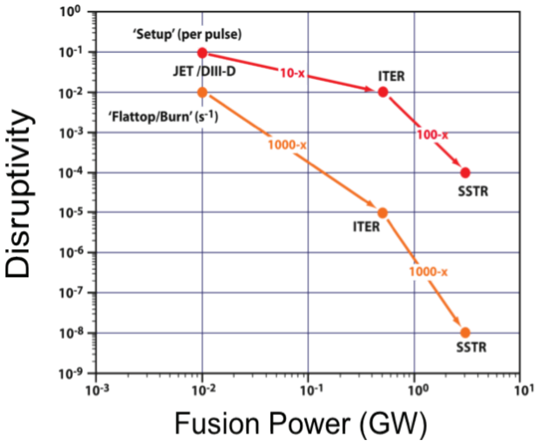


Figure 11. ITER will provide an important test of the ability to reduce the frequency of allowable disruptions ("disruptivity") in high-performance discharges, compared to that in current facilities. SSTR is a conceptual design for a steady-state tokamak reactor. (Figure reproduced from J. Wesley, "Disruptions: A Personal View," Research Needs Workshop white paper #18, http://burningplasma.org/web/renew_whitepapers_theme1.html.)

Disruption prediction

Accurate prediction of an impending disruption is a key element of both avoidance and mitigation of disruptions and has not been demonstrated to date. In particular, optimizing the reliability and accuracy of prediction (including minimization of "false-positive" results) will be critical. Research needs include:

- Real-time MHD calculations for assessment of the approach to stability limits.
- Accurate real-time profile diagnostics for input to MHD stability calculations.
- Diagnostics for direct detection of stability limits, e.g., real-time MHD spectroscopy to measure damping rates of large-scale MHD modes.

Development of real-time stability assessment and prediction, including diagnostics, stability calculations, and damping measurements, can be done in existing facilities and with low-power operation in ITER. Two key issues are the accuracy of these diagnostics and analysis techniques to minimize any loss in plasma performance and the use of diagnostics that are compatible with the stringent requirements of ITER and DEMO.

Disruption mitigation

Some disruptions are not predictable even in theory, such as those resulting from a sudden influx of impurities, failure of a power supply, etc. For these and other cases where avoidance fails, a preemptive rapid shutdown is required. Injection of sufficient quantities of gas or other impurities removes the plasma energy by radiation and raises the particle density to the level needed for collisional suppression of a runaway electron avalanche. Present experiments have demonstrated reduction of peak heat loads and electromagnetic forces through gas injection, but sufficient density for suppression of runaway electrons is a remaining challenge. Research needs in this area include:

- Development and evaluation of options for delivering material to the plasma and improved predictive capability for the physics of assimilating material into the core plasma.
- Assessment of radiation asymmetry during rapid shutdown and of the requirements for the number of injectors and their location.
- Improved understanding of the confinement and loss of runaway electrons, and development of techniques to suppress or control the position of the runaway electron beam.

Much development of rapid shutdown techniques can be done in existing facilities. The exponential dependence of runaway electron generation on plasma current indicates a need to include high-current devices (JET and JT-60SA) in these studies. Accurate, validated modeling will be required to extrapolate these results to the quantitatively different size, current, and energy of ITER. Due to its large plasma current, ITER will significantly extend the studies of runaway electron generation and provide critical data for DEMO.

Disruption characterization and modeling

Greater use should also be made of present facilities, in ITER and DEMO-relevant plasma configurations and operating modes, to compile and interpret the "statistical" aspects of disruption causes and characteristics. The characteristics and consequences of "worst-case" disruptions in ITER and reactor tokamaks are already reasonably well known, but further data is needed on current quench and electromagnetic loads, attributes of thermal quench and effect on plasma facing components, runaway electron generation, confinement and loss, and effect of mitigation methods. As part of this effort, there is a need to develop theory-based, integrated, 2-D and 3-D models that can accurately predict thermal and electromagnetic loadings and the potential for runaway electron conversion during disruptions in ITER. The same models are also essential for interpreting how disruption mitigation actions will affect these quantities. These models must be validated with data from existing facilities.

Large ELM heat flux

Edge instabilities known as ELMs are a nearly ubiquitous feature of H-mode discharges. These instabilities are associated with the development of a large edge pressure gradient and cause very rapid release of significant amounts of energy into the open-field line region. Recent studies for ITER have concluded that an incident energy impulse of more than ~0.3% of the total thermal plasma energy, which corresponds to a loss of ~1 MJ per ELMs event, can cause tile fatigue and cracking as well as erosion, and larger energy losses can ablate or melt divertor materials, potentially degrading the purity of ITER plasmas and greatly reducing the lifetime of the ITER divertor. These results imply a need to reliably reduce the energy impulse by a factor of ~20 for the level expected for unmitigated ELMs in ITER.

The ITER team has adopted two strategies toward this goal: (1) ELM suppression by means of non-axisymmetric coils and (2) ELM size reduction with the use of pellet injection to induce more frequent ELMs. Because the core confinement in H-mode plasmas is strongly correlated with the parameters in the pedestal, this must be done without significantly degrading the pedestal conditions. The impact of unmitigated ELMs on solid divertor targets is so severe in both ITER and future higher-power long-duration fusion facilities that multiple approaches are being investigated to address this issue.

ELMs suppression

The application of non-axisymmetric fields in the edge region is a promising technique to suppress ELMs, as shown in experiments on DIII-D, which demonstrated conditions in which the ELMs were fully suppressed (see Figure 12). Further experimental and theoretical research is required to provide a firm scientific and technical basis to support the ITER program and extrapolate to DEMO by addressing the following topics:

- Determine the requirements for radial localization of the perturbed field and the magnetic spectrum, which appears to be important for the attainment of ELMs suppression.
- Understand the transport processes associated with these fields in the edge and pedestal region.

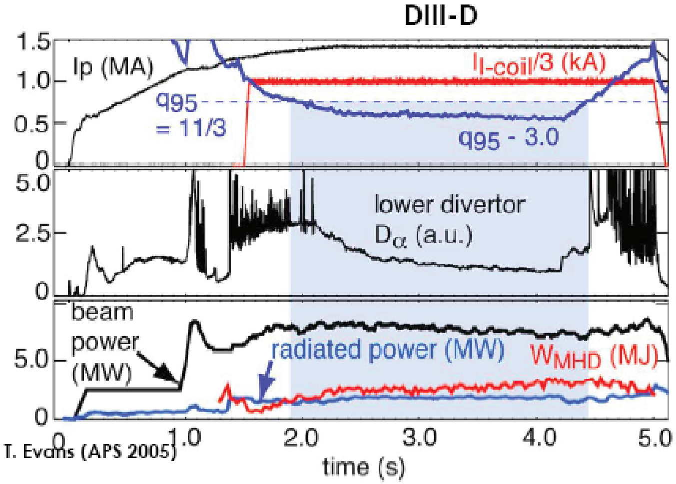


Figure 12. The application of non-axisymmetric fields in the plasma edge region, as indicated by the nonzero current I in the I-coil, is a promising technique to suppress ELMs, as shown in DIII-D experiments. (Figure courtesy of T.E. Evans; for a related article, see Nature Physics 2 [2006] 419.)

- Determine the perturbed field requirements to control ELMs over a broad range in q.
- Determine the requirements for avoiding deleterious nonresonant fields.
- Evaluate the compatibility of core pellet fueling with the application of non-axisymmetric fields.
- Develop magnetic coils that can operate inside the ITER vacuum vessel, where the coils are exposed to high neutron fluence and require remote maintenance.

ELM-free operational regimes

The development of operating regimes with the occurrence of beneficial edge instabilities to control the edge pressure gradient and avoid the onset of ELMs is an active area of research, resulting in the discovery of quiescent H-mode (QH) and enhanced D_α (EDA) modes of operation. An alternative strategy is to develop improved confinement regimes with a low-confinement mode (L-mode) edge. Research needs in this area include:

- Identify the underlying mechanisms responsible for the edge instabilities in the QH and EDA modes of operation and determine if these can be used on ITER.
- Identify tools to control the edge instabilities and pedestal parameters. The edge instabilities are needed to avoid the accumulation of impurities and density build-up in H-mode discharges, which are ELM-free.
- In improved confinement regimes with an L-mode edge, assess whether the lower edge pressure gradient is compatible with an increased peaking factor without degradation of the plasma ideal beta limit.

Operation with small ELMs

While the suppression of ELMs may be the preferred strategy for ITER and future machines that use an H-mode edge, it may be necessary to develop operating regimes with frequent small ELMs or adopt a strategy to make frequent small ELMs reliably. To reliably operate in such small-ELM regimes, it would be necessary to achieve reduction of the ELM energy loss fraction to $\Delta W_{ELM}/W < 0.3\%$ in ITER for tungsten or carbon fiber composite (CFC) divertor targets. Experimentally, the magnitude of the heat loss per ELM event scales roughly with the ELM period. Thus, increasing the ELM frequency enables the production of smaller ELMs. The research needs in this area include:

- Develop regimes of operation such as Type II, Grassy, or Type IV ELMs, which reliably meet the requirements for small ELMs in conditions that extrapolate to ITER.
- Assess whether techniques such as pellet pacing, vertical jogs, or application of $n > 0$ oscillating magnetic fields can reliably stimulate small ELMs.

The experiments described to control ELMs can be done on existing facilities with modest upgrades in the tools to suppress ELMs by the application of non-axisymmetric fields and the tools to modify the edge conditions.

Alpha particles ejected by plasma instabilities

The standard ITER Q~10 H-mode is expected to have up to ≈15 MJ of stored energy in the fusion-alpha population; lower density, higher temperature operating points may have more. Plasma instabilities can enhance the loss of the fusion alphas beyond the relatively small "first orbit" losses. A loss of 5% to 10% of the ≈15 MJ of confined alphas — comparable to loss fractions for fishbones or toroidal Alfvén eigenmode (TAE) avalanches in present devices — would correspond to 0.75 MJ to 1.5 MJ per event. This would be at the margin for reaching the surface-melting threshold if the area for heat loss is comparable to that of the background plasma.

Further, the lost alphas, with energies of up to 3.5 MeV, may present a danger to plasma facing components beyond that of transient heating. Laboratory experiments have found that the 3.5 MeV alphas can severely degrade metal target plates by implanting helium in the tungsten divertor targets. The deposition pattern for alpha particles lost during toroidal Alfvén eigenmode avalanches, sawteeth, edge localized modes, fishbones, or neoclassical tearing modes in the divertor or elsewhere has not been calculated for ITER, so neither the magnitude of the fluence nor the deposition area is known. Rough estimates of the alpha-particle loss due to instabilities in ITER indicate that blistering might degrade the lifetime of first wall and divertor components and contribute to dust production. Note that loss of the entire fusion alpha population in a single event, such as a disruption, is well below the blistering threshold.

Enhanced losses of energetic particles in present-day experiments seem to correlate both with instabilities driven by the background plasma such as tearing modes, ELMs, disruptions, sawteeth, and kinetic ballooning modes (KBM) as well as with energetic particle-driven instabilities such as TAEs and other related Alfvén eigenmodes, so-called energetic particle modes (EPM), and fishbones. Losses of up to 20% per event have been observed, with a timescale of ~1 ms. While substantial progress has been made in identifying the onset conditions for the occurrence of the fast ion losses, a quantitative prediction for the alpha loss rate and the distribution of the lost alphas is not yet available. Predicting the magnitude, frequency, and localization of the losses is important for the design of the plasma facing components. Likewise, it is necessary to understand the resilience of tungsten or other plasma facing components that are affected by lost alphas against the combination of high-transient power loads and high-alpha fluence. Research needs in this area include:

- Determine the thresholds for TAE avalanche, fishbone, and related EPM stability, and compare with linear and nonlinear theory, including gyrokinetic effects.
- Benchmark predictions from simulation codes of fast ion losses in the presence of MHD mode activity against experimental measurements of mode amplitudes, profiles and frequencies and measured fast ion losses and redistribution. This should be done for all forms of MHD instabilities that might cause fusion alpha losses.
- Measure fast ion deposition profiles on divertor tiles and other plasma facing components, and compare with the predictions from simulation codes.
- Improve the understanding of material damage due to alpha-particle bombardment.

Present experimental facilities, together with the application of new codes under development, could be used to develop a predictive capability for energetic particle losses in ITER. Test stand experiments are needed to study the effect of alpha particle bombardment.

DIAGNOSING A SELF-HEATED PLASMA

The ability to make accurate measurements on hot, magnetically confined plasmas has driven much of the understanding and progress to date in the fusion energy sciences field. Similarly, accurate measurements are required in ITER, not only to facilitate control of plasma quantities to achieve burning plasma conditions, but also to obtain information about burning plasma behavior that is important for future applications (see Figure 13). High-quality measurements on ITER will be more challenging than on present-day devices due to the neutron-rich environment in which diagnostics must operate, which will require the development of new diagnostic techniques. The deployment of these diagnostics will likely pace the development of the knowledge of burning plasma physics, for "The required progress in key areas will not be possible without a significant expansion of our diagnostic capabilities. Quite simply, we cannot understand what we cannot measure."[1]

CURRENT STATUS

Issues

As we proceed toward creating burning plasmas in ITER and, beyond that, toward creating *steady-state* high-performance burning plasmas, there are major challenges at each step for making the necessary measurements. In its report analyzing the issues and gaps for fusion research arising in a path to DEMO, the FESAC "Greenwald Panel" identified the need for measurements that will

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[1] *Plasma Science: Advancing Knowledge in the National Interest, National Academy Press (2007), p 120.*

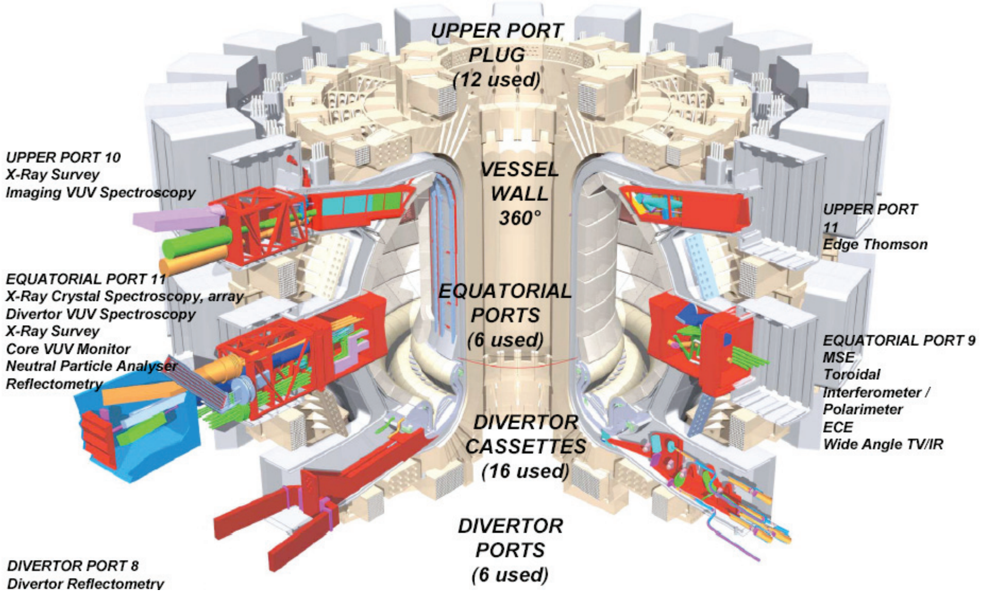

Figure 13. In ITER, about 40 large-scale diagnostic systems will be used for physics studies, plasma control, and device protection. (Figure courtesy of ITER Organization.)

facilitate the production of the steady-state burning plasmas. The panel said we must "…make advances in sensor hardware, procedures and algorithms for measurements of all necessary plasma quantities with sufficient coverage and accuracy needed for the scientific mission, especially plasma control."[2]

With these challenges motivating our planning and efforts, we organized the research opportunities for "diagnosing a self-heated plasma" into three areas:

- Development of the *measurement capability* needed for physics understanding of issues crucial in the achievement of burning plasmas.
- Development of the necessary *compatibility of the measurements, calibration strategies, and reliability requirements* necessary for the long-pulse nuclear environment.
- Development of the necessary degree of real-time *interpretation and analysis* of the measurements to facilitate the level of plasma control essential for device protection and advanced performance objectives.

These developmental needs apply to ITER burning plasmas (Theme 1) and also to high-performance steady-state burning plasmas (see the section on Measurements in the Theme 2 chapter).

Making the necessary measurements to achieve the desired predictive capability and required plasma control represents a major challenge and opportunity. Many of the diagnostics that have enabled such significant progress in fusion science to date will face formidable obstacles when applied to measurements in burning plasmas. For some existing techniques, performance will suffer

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[2] *Research Priorities, Gaps, and Opportunities, Panel Report for FESAC (2007), p. 28.*

due to incompatibility of the measurement physics with burning plasma parameters — e.g., large plasma size and high density may prevent the use of diagnostics that depend on penetration of neutral beams. Relativistic effects will significantly affect some diagnostic interpretations. Some new and difficult measurements must be identified and perfected to insure the safety of the plant (e.g., dust accounting), and others must be perfected for the safety of the device (e.g., sensors for disruption avoidance, tritium inventory, and first-wall ablation). However, the greatest challenge the diagnostics will face is the hostility of the near-plasma environment and the uncertainty of the impact of those environmental hazards. We note the following diagnostic concerns for ITER:

- No clear way to measure some desired quantities, e.g., dust and lost alpha particles.
- Hazards to diagnostics close to the device: thermal excursions, irradiation-induced damage and noise, degradation due to erosion/deposition, disruption-induced vibrations and component displacement. Lifetimes of first mirrors are a particular concern. Evidence from both modeling and experiments indicates that severe problems may exist for ITER due to surface deposition and erosion, but understanding is limited and uncertainty is large. ITER will require difficult trade-off decisions between measurement performance and reliability, without well-founded guidance.
- Understanding of environmental impact to diagnostics is lacking in many areas, especially for combined hazards.
- Lifetime and performance of in-vessel sensors, cables, and connectors in the nuclear environment.
- Ability to maintain calibrations of instrumentation over long pulses.

To highlight the research needs of measurements and diagnostics for burning plasmas, we assessed their urgency and magnitude, and constructed Table 1. This table also serves to illustrate the magnitude of the steps required when moving from present-day diagnostic capabilities, through ITER, to a DEMO device.

🟩	Mostly OK
🟨	Some R&D needed
🟥	Significant R&D needed

	Measurement readiness	Control capable	Compatibility	Reliability	Real-time interpretation
Sufficient for present-day tokamaks	🟨🟩🟨🟩	🟩	🟩	🟨	🟩🟨🟩🟨
Should be sufficient for ITER	🟥	🟨	🟥	🟥	🟨
Should be sufficient for a DEMO	🟥	🟥	🟥	🟥	🟥

Table 1. Assessment of Measurement Research Needs. Note that the "Measurement Capability" sub-issue has been split here into the "measurement readiness" and "control capable" columns. Elaboration of the assessments of the "measurement readiness" and "real-time interpretation" for present-day tokamaks is necessary. Present-day tokamaks have excellent diagnostic capability. However, additional research and development on current devices is needed in those categories to gain crucial understanding in specific areas and to perform crucial validation experiments in support of future needs (see text); hence the "striped" assessment for those areas.

Recent Accomplishments and Progress

Accurate measurements of hot, magnetically confined plasmas have had a profound effect upon our ability to understand and manipulate those plasmas. Great progress has been made during the past 20 years in measuring crucial quantities of such a complex and strongly interacting system. We point out only a few examples here, selected for their relevance to this Theme.

- Plasma shape has been shown to have a profound effect upon stability and confinement. We have developed the capability to measure the shape of the flux surfaces with an accuracy of a few percent of the plasma minor radius, i.e., roughly 1 cm.
- Knowledge of the profiles of fundamental quantities like density, temperature, and current density is crucial for evaluating fusion power production, transport, and heating efficiency. The gradients of these quantities also provide crucial input to comparisons with theory. We now make profile measurements of (electron) temperature and density with a spatial resolution in the plasma edge approaching the ion gyroradius, roughly 1 mm. (The importance of current density profile measurements is also discussed in the "Accomplishments" section of the Theme 2 chapter.)
- The diagnosis of fast ion instabilities is especially important for burning plasmas, since alpha heating is crucial for achieving the burning plasma state. Great progress has been made in measuring the confined-alpha particles, and the signatures and effects of a number of fast ion instabilities with exotic names like "fishbones," "Alfvén eigenmodes," and "energetic particle modes."
- Plasma rotation and rotation shear play an important, though not yet fully understood, role in plasma confinement. Plasmas exhibit significant "intrinsic" rotation, i.e., they rotate even in the absence of any known external momentum input. Measurements of plasma rotation and rotation shear have demonstrated the importance of these phenomena and stimulated theory to understand them.

SCIENCE CHALLENGES, OPPORTUNITIES, AND RESEARCH NEEDS

Development of the measurement capabilities needed for physics understanding of issues crucial for the achievement of burning plasmas

One opportunity in this area is to provide targeted measurement capability sufficient for physics understanding in a number of critical areas, with the ultimate goal being to develop a predictive capability that can be confidently extrapolated to future burning plasmas. The testing of comprehensive models of plasma behavior comprises a significant component of modern experimental programs. To date, model validation efforts in magnetic fusion plasmas have necessarily been limited in scope and have not been held to the rigorous standards employed in fields such as fluid dynamics and aerodynamics. This situation is evolving as more sophisticated and comprehensive approaches to validation are developed, including methodologies and metrics by which to evaluate progress and quantify the agreement of models with experiments.[3] Models of transport, macro-stability, energetic particle-driven modes, pedestal dynamics, edge physics, and scrape-off layer phenomena have advanced greatly in recent years, as computational capabilities, advanced numerical techniques, and understanding of the physics have matured. More comprehensive di-

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[3] *P. Terry et al., Phys. Plasmas 15 (2008) 062503.*

agnostic measurements of critical parameters (including details of equilibrium profiles, as well as the fluctuations and fluctuation-induced fluxes associated with various instabilities) and associated synthetic diagnostics have also become available, enabling more direct and detailed tests of model predictions. Further progress will require the development of measurement capabilities that are not currently available.

Thus, we need to develop and deploy on available devices the diagnostics necessary for physics understanding of critical issues for the achievement of burning plasmas. The measurements would be used for model validation and predictive capability. It is important to keep in mind that motivation for timely deployment of such diagnostics comes from the knowledge that non-burning devices may provide the best opportunity for these difficult measurements, since the difficulty of making measurements is greatly increased on burning plasmas. Experience has shown that it takes about ten years to develop a new diagnostic concept into a reliable full-profile, full-temporal capability for existing devices.

Other Panels across all of the ReNeW Themes have identified measurement needs for critical understanding. Examples are:

- "Initiate DEMO PFC diagnostics development in laboratories and deployment for both solid and liquid surface options. Perform integrated testing in operating tokamaks and provide data on the qualification of materials." (Theme 3 PFC Thrust)
- "Development of new advanced diagnostics is an essential component for validation." (Theme 1 Alpha Particle Physics Panel)
- "Measure turbulence to fullest possible extent for thermal transport validation." (Theme 1 Extending Confinement Panel)
- "Actuators and diagnostics – general control research requirements identified common to most control topical areas." (Themes 1 and 2 Joint Panel on Control)
- "Diagnostics to detect stability limits" and "Real-time profile diagnostics for input to MHD stability." (Themes 1 and 2 Joint Panel on Off-Normal/Transient Events)
- "Experiments: Improve diagnostics on existing devices." (Theme 2 Modeling Panel)

We provide only a sampling of examples here, advocating that a process and organization be applied to insure that the critical issues are identified, and that the appropriate diagnostics and simulations to be validated are developed. Additional criteria for selecting such measurements should be based on the weight they have in discerning models and on the priority that validating models have, for example, on public and facility safety, operations, and fusion performance optimization.

Development of the necessary compatibility of the measurements, calibration strategies, and reliability requirements

The ITER environment brings new challenges and risks for measurements and diagnostics. We propose research to enhance the capability, compatibility, calibration strategies, and reliability of ITER diagnostic measurements. The work plan of this research would be driven by the establishment of a program that, in collaboration with the ITER team, would continually reevaluate the physics and engineering measurement requirements in ITER as they change in response to evolving understanding and capability. Research needs would be identified when the ITER measurements requirements are judged to be at high risk of being unmet. In combination with this on-going assessment, we also identify these specific research needs:

- Research on and development of new techniques to address existing measurement gaps, e.g., dust, lost alpha particles, edge flows.
- Research to reduce risk on the measurement capabilities of planned techniques, due to the ITER environment. The primary example of such a risk is the first mirror, for which there are concerns about lifetime and stability. There are also uncertainties in the estimates of lifetimes and performance of in-vessel sensors, cables, and connectors that are too large. These issues connect to many independent diagnostic systems.
- Research to develop new techniques that use robust plasma front end interfaces, e.g., microwave and X-ray techniques.
- Research to develop calibration strategies and reliability requirements.

Development of the necessary degree of real-time interpretation and analysis of measurements

In this area we identify the following research needs.

Develop real-time measurement and analysis capability

Reactor-relevant burning plasmas will operate for long times, and measurements will need to be available continuously to optimize performance. Control systems will require the availability of real-time measurements to control actuator systems. Significant progress in this area has allowed for real-time measurement of plasma profiles and equilibria in present devices. These techniques will need to be significantly advanced, expanded, and tested for application to burning plasmas. Performance optimization requires real-time adjustment of temperature, density, and current density profiles. Different operational scenarios have different real-time measurement and feedback requirements. For example, advanced scenarios (with high beta and high bootstrap fraction) will likely be most demanding in real-time analysis.

This research need is to be coordinated with integrated Plasma Control research.

Develop strategies and techniques for prediction, rapid evaluation, decision for prevention, mitigation of disruptions/off-normal events required for machine operation and protection

Any long-pulse plasma experiment will need to confront off-normal events such as disruptions, which can ablate and melt first-wall materials, drive large halo currents, create runaway electrons, and put stress on structural components. Numerous techniques are being developed to mitigate the adverse effects of disruptions. Implementing such systems on burning plasma experiments will be a special challenge to the interpretation and analysis of measurements. Equally important is the necessity to prevent off-normal events from occurring in the first place. This requires assessment of when potentially dangerous conditions (pressure, gradients, position, etc.) are occurring and the implementation of prevention methods (e.g., reduced power input, plasma movement). Diagnostic systems will need to detect when dangerous conditions are present or being approached, and intelligent analysis algorithms will need to make real-time decisions. The detection/decision process must occur on the time scale of milliseconds to actuate a prevention and/or mitigation strategy. This must be done with high reliability, since the safe operation and integrity of the device may be at risk from the off-normal event. Development and testing of such systems is crucial to the success of burning plasma systems.

Develop methods of Integrated Data Analysis and continue implementation of synthetic diagnostics in relevant codes

Optimizing the interpretation and analysis of measurements aids in more complete exploitation of data for understanding and predicting burning plasmas. We identify two areas where continued development is needed and will provide enhanced measurement capability. In particular, methods of "integrated data analysis" are required to combine related measurements from complementary diagnostics to minimize errors, resolve data inconsistencies, and obtain a validated data set.[4] Because the implementation of "synthetic diagnostics" (i.e., predicting the quantity that a diagnostic actually measures) in modeling codes has been shown to be useful, more accurate diagnostic modeling within our available relevant codes is highly desired.

CONCLUSION

Time Scales for Research

We have deliberately not called for research on *specific* measurement techniques and diagnostics, since in many cases we cannot at present define the specific techniques that will be used or desired – indeed the creation of capable and compatible techniques is a major goal of the research. As the research program defines the desired measurement capabilities and thereafter, it must *assess* the existing capability of making each measurement and *facilitate* the research and development still needed. This will certainly require responsive support for short- and long-term development and implementation of *new* diagnostics and diagnostic techniques. Measurement capability for both burning and non-burning plasmas must be systematically developed. New diagnostic techniques can take ~10 years to mature. They must move from prototypes, to validation, and to reliability demonstrations on nonburning plasmas. Additionally, they must be modeled and/or tested (if possible) in a nuclear environment before they are commissioned on a burning plasma device.

Research Benefits

The quality and coverage of measurements directly affect essentially all aspects of experimental burning plasma science. This research will enable both predictive capability applied to burning plasmas, as well as increase the quality and number of the measurements used for burning plasma diagnosis and control.

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[4] *See, for example, R. Fischer et al., Plasma Phys. Control. Fusion 45 (2003) 1095–111.*

Correspondence to Research Thrusts

The research requirements for this Theme ("Understanding and Achieving the Burning Plasma State in ITER") are to be addressed by several priority research directions, called Research Thrusts. These will be described in detail in Part II of this Report.

In the table below, we summarize how the research requirements and opportunities for the six panels of Theme 1 map into various Research Thrusts.

PANEL	**RESEARCH THRUSTS**
Understanding Alpha Particle Effects	Thrust 3: Understand the role of alpha particles in burning plasmas.
Extending Confinement to Reactor Conditions	Thrust 4: Qualify operational scenarios and the supporting physics basis for ITER.
Creating a Self-Heated Plasma	Thrust 4 (see above).
Controlling and Sustaining a Self-Heated Plasma	Thrust 5: Expand the limits for controlling and sustaining fusion plasmas.
Mitigating Transient Events in a Self-Heated Plasma	Thrust 2: Control transient events in burning plasmas. (also Thrust 3 [see above] for transient alpha-burst events)
Diagnosing a Self-Heated Plasma	Thrust 1: Develop measurement techniques to understand and control burning plasmas.

SUGGESTIONS FOR FURTHER READING

1. *Planning for US Fusion Community Participation in the ITER Program*. Prepared by the Energy Policy Act Task Group of the US Burning Plasma Organization (June 7, 2006). http://burningplasma.org/web/ReNeW/EPAct_final_June09.pdf
2. *A Review of the DOE Plan for US Fusion Community Participation in the ITER Program*. Prepared by the National Research Council (National Academy Press, 2009).
3. *ITER Physics Basis*, Nuclear Fusion, vol. 39, no. 12, pp. 2137-2664 (Dec 1999).
4. *Progress in the ITER Physics Basis,* Nuclear Fusion, vol. 47, no. 6, pp. S1-S413 (June 2007).
5. R. Hawryluk, *Results from Deuterium-Tritium Tokamak Confinement Experiments*, Rev. Mod. Phys. vol. 70, p. 537 (1998).
6. J. Jacquinot and the JET Team, *Deuterium-Tritium Operation in Magnetic Confinement Experiments: Results and Underlying Physics*, Plasma Phys. Control. Fusion, vol. 41, pp. A13-A46 (1999).
7. *Final Report–Workshop on Burning Plasma Science: Exploring the Fusion Science Frontier* (Fusion Energy Sciences Advisory Committee, 2000) http://fire.pppl.gov/ufa_bp_wkshp.html
8. *A Burning Plasma Program Strategy to Advance Fusion Energy* (Fusion Energy Sciences Advisory Committee, September 2002) http://www.ofes.fusion.doe.gov/More_HTML/FESAC/Austinfinalfull.pdf.
9. *Burning Plasma: Bringing a Star to Earth* (National Academy of Science, 2004).

BURNING PLASMAS IN ITER
THEME MEMBERS

UNDERSTANDING ALPHA PARTICLE EFFECTS

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HERBERT L. BERK, The University of Texas at Austin
NIKOLAI GORELENKOV, Princeton Plasma Physics Laboratory
RAFFI NAZIKIAN, Princeton Plasma Physics Laboratory
MICHAEL VAN ZEELAND, General Atomics

EXTENDING CONFINEMENT TO REACTOR CONDITIONS

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JERRY HUGHES, Massachusetts Institute of Technology
DAVID MIKKELSEN, Princeton Plasma Physics Laboratory
JOHN RICE, Massachusetts Institute of Technology
WILLIAM L. ROWAN, The University of Texas at Austin
PAUL TERRY, University of Wisconsin–Madison

CREATING A SELF-HEATED PLASMA

RONALD R. PARKER, Massachusetts Institute of Technology (Panel Leader)
LARRY BAYLOR, Oak Ridge National Laboratory
CHARLES KESSEL, Princeton Plasma Physics Laboratory
DALE MEADE, Fusion Innovation Research and Energy
MASANORI MURAKAMI, Oak Ridge National Laboratory
RON PRATER, General Atomics
STEVE WUKITCH, Massachusetts Institute of Technology

CONTROLLING AND SUSTAINING A SELF-HEATED PLASMA

(Joint with Theme 2 panel "CONTROL")

DAVID HUMPHREYS, General Atomics (Panel Leader)
JOHN FERRON, General Atomics (Panel Deputy Leader)
TOM CASPER, Lawrence Livermore National Laboratory
DAVID GATES, Princeton Plasma Physics Laboratory
ROB LAHAYE, General Atomics
TOM PETRIE, General Atomics
HOLGER REIMERDES, Columbia University
ALAN TURNBULL, General Atomics
MIKE WALKER, General Atomics
THOMAS WEAVER, Boeing
STEVE WOLFE, Massachusetts Institute of Technology

MITIGATING TRANSIENT EVENTS IN A SELF-HEATED PLASMA

(Joint with Theme 2 panel "OFF-NORMAL EVENTS")

RICHARD HAWRYLUK, Princeton Plasma Physics Laboratory (Panel Leader)
STEVE KNOWLTON, Auburn University (Panel Deputy Leader)
JON MENARD, Princeton Plasma Physics Laboratory (Panel Deputy Leader)
ALLEN BOOZER, Columbia University
ERIC FREDRICKSON, Princeton Plasma Physics Laboratory
BOB GRANETZ, Massachusetts Institute of Technology
VALERIE IZZO, University of California, San Diego
EDWARD J. STRAIT, General Atomics
JOHN WESLEY, General Atomics
DENNIS WHYTE, Massachusetts Institute of Technology

DIAGNOSING A SELF-HEATED PLASMA

(Joint with Theme 2 panel "MEASUREMENTS")

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DAVID BROWER, University of California, Los Angeles
DANIEL DEN HARTOG, University of Wisconsin–Madison
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RESEARCH THRUST COORDINATORS

Thrust 1 – Develop new measurement techniques to understand and control burning plasmas
DAVID W. JOHNSON, Princeton Plasma Physics Laboratory
Thrust 2 – Control transient events in burning plasmas
EDWARD J. STRAIT, General Atomics
Thrust 3 – Understand the role of alpha particles in burning plasmas
DONALD A. SPONG, Oak Ridge National Laboratory
Thrust 4 – Qualify operational scenarios and the supporting physics basis for ITER
RONALD R. PARKER, Massachusetts Institute of Technology
C. CRAIG PETTY, General Atomics

THEME LEADERS

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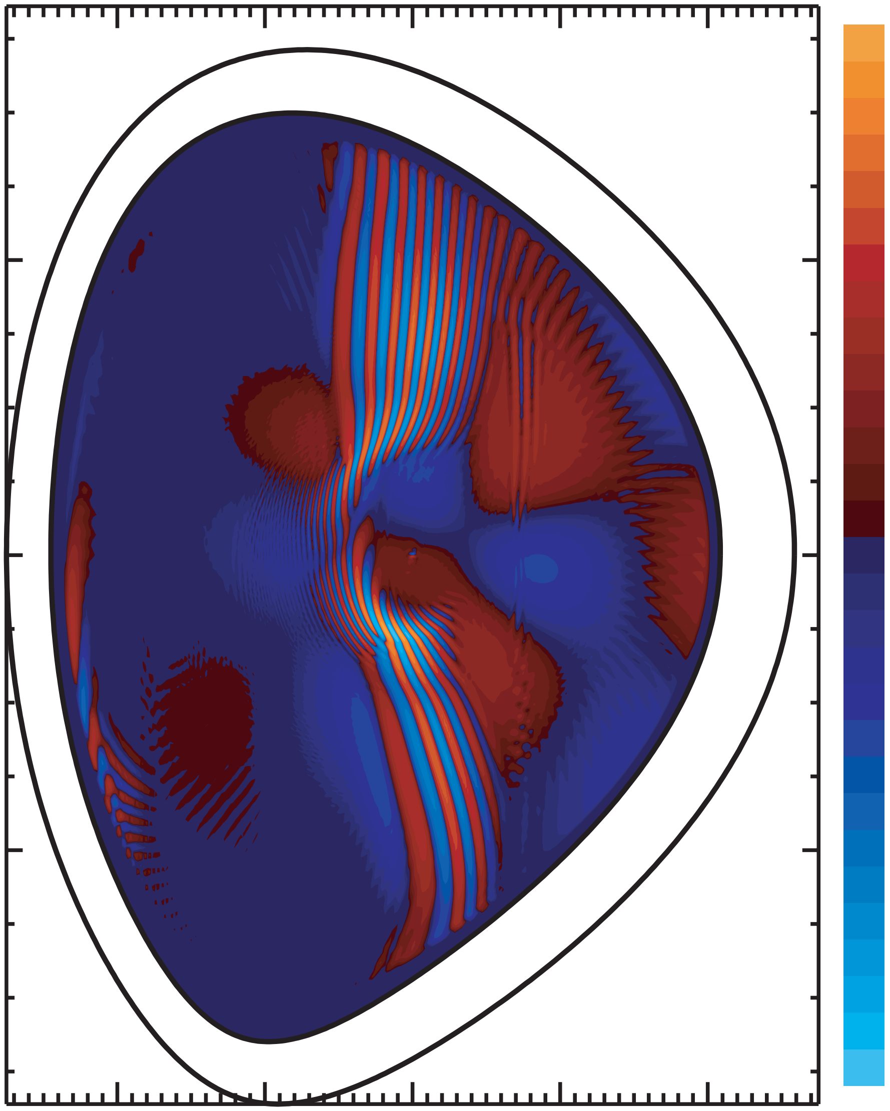
JOHN MANDREKAS, Office of Fusion Energy Sciences, U.S. Department of Energy

THEME 2:

CREATING PREDICTABLE, HIGH-PERFORMANCE, STEADY-STATE PLASMAS



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Nonlinear simulation of mode conversion of applied radiofrequency waves to shorter-wavelength modes in a tokamak. (From P.T. Bonoli, MIT)

THEME 2: CREATING PREDICTABLE, HIGH-PERFORMANCE, STEADY-STATE PLASMAS

Introduction

SCOPE AND FOCUS

This Theme corresponds directly to Theme A of the Priorities, Gaps and Opportunities Report[1], a key resource document for ReNeW. Its overall goal, as noted in that report, is, *"The state of knowledge must be sufficient for the construction, with high confidence, of a device that permits the creation of sustained plasmas that meet simultaneously, all the conditions required for practical production of fusion energy."*

This is an exceptionally broad and challenging goal, encompassing a large fraction of the current effort in the US Magnetic Fusion Energy Sciences program. As described in the preceding chapter on Theme 1, ITER will be an enormously important and necessary step toward obtaining this knowledge base, enabling for the first time the production and study of burning plasmas, in which more energy is produced by fusion than is used to heat them. However, it must be recognized that ITER is not designed to be a prototype for "practical production of fusion energy." Such a demonstration fusion reactor, often referred to as DEMO, to lead to economic and attractive fusion energy, would need advances in several key areas. Two in particular are the foci of Theme 2, as represented by its title:

High fusion power density: The needed system size for a given fusion power output is determined by the fusion power density P_{fus} in the burning core. This is the main measure of "*high performance*." P_{fus} is directly proportional to the square of the plasma pressure, $p^2 \propto n^2 T_i^2$, where n is the density and T_i the ion temperature. ITER is designed to achieve near optimal temperatures for fusion cross-sections; hence, the main avenue for improvement is increasing the plasma density. The fusion power is also proportional to $\beta^2 B^4$, where $\beta=2\mu_0 p/B^2$ is the plasma pressure normalized to magnetic pressure, and B is the magnetic field. Improvements can thus be realized by increasing β, which is limited by plasma stability and dependent on magnetic configuration, and/or by increasing magnetic field B.

Steady-state: Rather than operating in a pulsed mode, driven inductively by a transformer, a fusion device should be *sustained* for indefinite durations, with high reliability. For tokamaks, this means current must be provided and sustained by a combination of internally generated "bootstrap" current (driven by gradients in the density and temperature profiles) and externally driven currents (using microwaves or neutral beams). Other magnetic configurations, such as stellarators, sustain plasmas without large current, through 3-D (non-axisymmetric) magnetic fields. In order not to use too large a fraction of the electricity produced for plasma sustainment, the amount of required external heating and current drive must be modest.

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[1] *"Priorities, Gaps and Opportunities: Towards A Long-Range Strategic Plan For Magnetic Fusion Energy" Report to FESAC Oct 2007, chaired by M. Greenwald.*

Increasing power density and pulse length also increases the difficulty of handling the heat and particle fluxes out of the plasma and their impact on the first wall, the emphasis of Theme 3. Theme 4 addresses many other challenges that must be met for practical fusion energy.

Equally important in acquiring the "sufficient state of knowledge," "with high confidence" is that the behavior of fusion plasmas and their associated systems be **predictable**. Progress in scientific understanding and predictive capability will enable confident extrapolation from present and planned experiments, including ITER, to next-step devices and DEMO.

The work of Theme 2 was divided into seven panels, each focused on one of the issues identified in the resource document, and representing a wide range of expertise in both physics and engineering science. While these topics are distinct, progress in each is crucial for success in the overall goal. The panels, FESAC issues, and role in this Theme are outlined briefly below. Key research requirements identified by each panel are detailed in this Chapter.

> **1. Measurement.** *Make advances in sensor hardware, procedures and algorithms for measurements of all necessary plasma quantities with sufficient coverage and accuracy needed for the scientific mission, especially plasma control.*
>
> Measurements are crucial both for obtaining the needed scientific understanding, and for creating and controlling steady-state, high-performance plasmas. They will be much more challenging in a fusion environment than in present experiments.
>
> **2. Integration of steady-state, high-performance burning plasmas.** *Create and conduct research, on a routine basis, of high-performance core, edge and scrape-off layer (SOL) plasmas in steady state with the combined performance characteristics required for DEMO.*
>
> Demonstration and study of sustained plasmas in the high-performance regimes envisaged for practical fusion energy will clearly be required. This will involve integrating both plasma parameters, and the tools and results of other panels.
>
> **3. Validated theory and predictive modeling.** *Through developments in theory and modeling, and careful comparison with experiments, develop a set of computational models that are capable of predicting all important plasma behavior in the regimes and geometries relevant for practical fusion energy.*
>
> The predictive capability embodied in a comprehensive set of well-validated models will guide the experimental demonstrations and represent the plasma physics knowledge base for proceeding to develop fusion energy.
>
> **4. Control.** *Investigate and establish schemes for maintaining high-performance burning plasmas at a desired, multivariate operating point with a specified accuracy for long periods, without disruption or other major excursions.*
>
> Maintaining plasmas in high-performance states, which may be above passive stability limits, requires many plasma parameters and profiles, with myriad interactions, to be controlled simultaneously. This needs to be accomplished with higher reliability and more limited external power than in present experiments.

5. Transient plasma events. *Understand the underlying physics and control of high-performance magnetically confined plasmas sufficiently so that "off-normal" plasma operation, which could cause catastrophic failure of internal components, can be avoided with high reliability and/or develop approaches that allow the devices to tolerate some number or frequency of these events.*

As plasma energies increase, transient events, which range from periodic edge instabilities (i.e., edge localized modes — ELMs) to sudden loss of plasma current (disruptions), will become a more serious problem for fusion systems, with the potential to disrupt and damage the device. Preventing or managing them is thus a critical requirement.

6. Plasma modification by auxiliary systems. *Establish the physics and engineering science of auxiliary systems that can provide power, particles, current and rotation at the appropriate locations in the plasma at the appropriate intensity.*

Heating, fueling, sustainment and control of fusion plasmas must be carried out by a set of external systems. Examples are heating and current drive systems using waves at various frequencies or high-energy beams, and frozen pellets for core fueling. The high density and pressure, and harsh environment, of a DEMO will be much more challenging for such systems than present experiments.

7. Magnets. *Understand the engineering and materials science needed to provide economic, robust, reliable, maintainable magnets for plasma confinement, stability and control.*

Magnets are integral components of all magnetic fusion energy facilities, used for containing hot plasmas and for all aspects of operation. High-performance superconducting magnets will be needed for steady-state operation. Advances in magnet materials and components, such as those offered by new high-temperature superconductors, could potentially increase B and lead to fusion systems that are more attractive in many other respects.

A key distinction between the assessment being carried out by ReNeW and the work of the Priorities, Gaps and Opportunities Panel is that the latter primarily assessed gaps that would remain following the successful completion of ITER and presently planned research. The research activities identified by ReNeW start from the present state of knowledge. For many issues, work is needed as part of preparation for successful burning plasmas on ITER, while further progress on the same areas will be needed beyond ITER. There is thus considerable overlap in topics and expertise between Themes 1 and 2.

In recognition of this, three panels (Measurement, Control, and Transient Plasma Events) were organized jointly with Theme 1. To avoid duplication, these panels first assessed research requirements for Theme 1, found in the previous chapter, and then additional needs for predictable, high-performance, steady-state plasmas, found later in this Chapter. The combined two sections give a more complete presentation of the full range of issues and research requirements on those topics.

HIGHLIGHTS OF ACCOMPLISHMENTS

While the emphasis of ReNeW is on issues not yet understood, and requirements remaining to be met, it is important to realize that each of these topical areas is already the subject of considerable research, and that impressive progress has been made. In this section we provide a few examples of recent accomplishments that have advanced research toward the goal of "predictable, high-performance, steady-state plasmas." Many more can be found in the scientific literature, including excellent review papers, and in recent community reports. Some suggestions for further reading are given at the end of this Chapter.

Accurate **measurements** of hot magnetically-confined plasmas have had a profound effect upon our ability to understand and manipulate those plasmas. This in turn has enabled the development of attractive operational regimes. Measurement of the local magnetic field inside a tokamak plasma, which was not possible until the 1990s, provides information on the current density profile. This is critical to understanding plasma confinement; current-profile control is one of the cornerstones on which advanced tokamak scenarios are based. Measurements of fluctuations and fluctuation-induced fluxes, associated with various instabilities as well as turbulence and turbulent transport, have provided direct quantitative evaluation of some of the sources of anomalous transport in plasmas. A powerful suite of diagnostic techniques has been developed over the years, including active spectroscopy, reflectometry, various kinds of imaging, and magnetics. Even more capability in this area is desired to measure other fluctuating quantities over shorter wavelengths. For steady-state or very long-pulse plasmas, the interactions between the plasma and the plasma facing wall, and the characteristics of the wall, become crucial. This is due to the long "wall-equilibration" time scale for such processes as particle retention and deposition, as well as erosion and ablation of material. Our understanding of various wall materials and of these interactions has increased enormously over the past 20 years, primarily as a result of the measurement capability that has been developed.

Impressive advances in plasma **control,** including **avoiding transient events,** have also been achieved, in part due to the improvements in measurements. These accelerated substantially in the 1990s with the advent of control-intensive advanced tokamak and related regimes in highly shaped, diverted, high-β plasmas operating routinely near or beyond magnetohydrodynamic (MHD) stability limits. Some key achievements worldwide over the last decade include:

- Complex model-based multivariable axisymmetric shape and stability control, giving routine actively stabilized operation of highly-elongated tokamak plasmas within 2% of the ideal limit for the axisymmetric MHD instability.
- Suppression of non-axisymmetric resistive MHD modes (including resistive wall modes, by applying non-axisymmetric fields, and neoclassical tearing modes, using electron cyclotron current drive). enabling operation significantly beyond no-wall stability limits.
- Suppression of edge localized modes by the application of non-axisymmetric fields.
- Multi-point regulation of current profiles using multiple combined actuators.
- Control of formation and subsequent regulation of internal transport barriers.
- Simultaneous control of divertor heat flux and total energy confinement by varying fueling, pumping, and impurity injection.

Enormous strides have been made in **theory and predictive modeling**, enabled by advances in massively parallel computing, in concert with a strong basic theory effort, and **validation** with new and detailed measurements. In the area of **auxiliary systems**, much of the complex interaction of radiofrequency waves with core plasmas, leading to heating and current drive, is now understood and can be accurately modeled. As one example of many during the past decade, significant improvements have been made in simulating wave-particle interactions in the ion cyclotron range of frequencies (ICRF). It is now possible to simulate ICRF mode conversion to ion Bernstein wave (IBW) and ion cyclotron wave (ICW), which have wavelengths much shorter than that of the incident fast magnetosonic wave (Figure 1). A short wavelength mode detected using a Phase Contrast Imaging technique on Alcator C-Mod was first expected to be an IBW, but appeared at the wrong wave number and spatial position. Theory and simulation, including comparison with a synthetic diagnostic in the code, led to the discovery of the first observation of an ion cyclotron wave. Very recently, the use of spectral full-wave solvers in the lower hybrid (LH) range of frequencies has made it possible to study focusing and diffraction effects in this regime. Accurate prediction of electron cyclotron resonance heating and current drive has been carried out and validated extensively on the DIII-D tokamak. Validation of ion-scale turbulence and transport models with new turbulence diagnostics has similarly increased our understanding in the transport area.

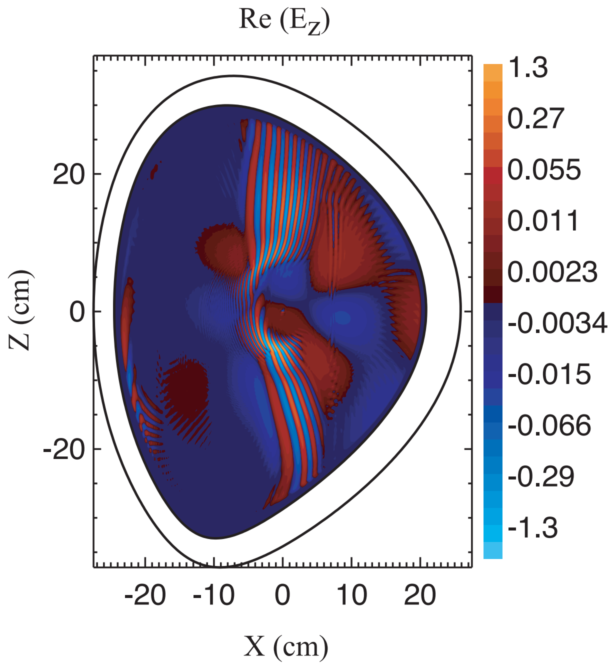

Figure 1. Simulation of mode conversion of incident ICRF waves to shorter wavelength modes with the TORIC full wave code. From J.C. Wright et al, Phys. Plasmas, 11(5) (2004) 2473.

Building on the measurement, prediction and control tools described above, Integrated Scenarios have been produced on many tokamaks that achieve many of the normalized parameters envisaged for attractive and economic tokamak reactors. In advanced tokamaks such as DIII-D in the US, and JT60-U in Japan, fully noninductive discharges have been produced with neutral beam and electron cyclotron current drive, and about 60% bootstrap current. In other discharges, up to 100% bootstrap fraction has been obtained. Normalized pressures well above passive stability limits have been maintained. Integrated simulation capability of all scenarios has greatly improved. The ARIES-RS[2] and ARIES-AT[3] design studies, which project to competitively low costs of electricity, are based largely on these positive results. Key remaining challenges are to demonstrate that advanced scenarios can be extended to steady state, with needed reliability and at DEMO-relevant parameters, and to integrate them with relevant edge parameters and divertor solutions.

Progress has also been made in the area of **superconducting magnets**. The US played a key role in developing and testing magnets for ITER, notably the Central Solenoid Model Coil, the world's most powerful pulsed superconducting magnet (Figure 2). An example of strong international collaboration, its inner module was fabricated by a collaboration of US universities, laboratories, and industry, and the outer module was fabricated in Japan by a laboratory-industry collaboration. Joint tests in Japan achieved a magnetic field of 13 tesla, with a stored energy of 640 MJ at a current of 46,000 amperes and high stability to transients. The levitation coil of the Levitated Dipole Experiment (LDX), an innovative confinement device at a much smaller scale, was fabricated using new high-temperature superconductor material.

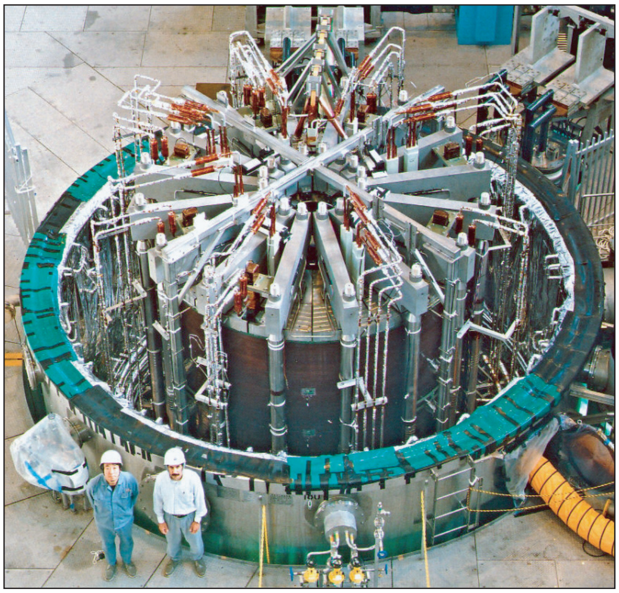

Figure 2. The Central Solenoid Model Coil, used to develop and test superconducting magnets for ITER. US engineers fabricated the Inner Coil Module and other key components. Photo courtesy of JAEA.

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[2] *F. Najmabadi and the Aries Team, "Overview of ARIES-RS tokamak fusion power plant," Fusion Engineering and Design 41 (1998) 365-370, and other articles in this issue.*

[3] *F. Najmabadi and the Aries Team, "The ARIES-AT advanced tokamak, Advanced technology fusion power plant," Fusion Engineering and Design 80 (2006) 3-23, and other articles in this issue.*

Research Requirements

In this section, a brief review of major issues and gaps, identified by the Priorities, Gaps and Opportunities Panel and by ReNeW panels, is given for each panel. Most importantly, the research required to address these challenges is assessed, both in terms of the quantitative targets that should be met, and the set of priority activities and tools – theory, computational, experimental and technological – that will be needed. These requirements form the basis of ReNeW Thrusts, as briefly outlined later in this Chapter and described in more detail in Part II of the Report.

MEASUREMENT: ADDITIONAL RESEARCH REQUIREMENTS

Overall Goal: *Make advances in sensor hardware, procedures and algorithms for measurements of all necessary plasma quantities with sufficient coverage and accuracy needed for the scientific mission, especially plasma control.*

The ability to make accurate measurements on hot magnetically confined plasmas has driven much of the understanding and progress to date in this field. Accurate measurements are presently routinely used both for improving our basic understanding of plasma behavior and for facilitating control of those plasma quantities for which control "actuators" exist. As we proceed *beyond* ITER, toward creating steady-state high-performance burning plasmas, there are *additional* major challenges for making the necessary measurements. (Please refer to the extended discussion of research opportunities for measurements in Chapter 1. Since the motivations for measurement capability are the same for these two Themes in many cases, we will not repeat the shared motivations here.) Beyond ITER, the environmental hazards for diagnostics close to steady-state burning plasmas are *much* greater and more daunting than for the ITER plasmas, e.g.,

- Two orders of magnitude higher fluence than ITER; flux 3-4 x that of ITER.
- Higher wall temperature: up to 650° C vs 240° C.
- Possibility of liquid metal walls.

We can better appreciate the increased challenges when we estimate the lifetimes of various components and diagnostics in such an environment. If we consider only the radiation-induced effects on components, place them in the equivalent locations as in ITER, and use present-day technology, then we find that the component lifetimes in a full-power DEMO-like environment are predicted to be:

- ~ 13 weeks for magnetic sensors.
- ~ 1 week for bolometers.
- ~ a few hours for vacuum ultraviolet (VUV) windows.
- ~ a few hours for pressure gauges.

These lifetimes are unacceptable, and significant development and careful design are required so that the necessary measurement capability is maintained for these steady-state, more self-organized, burning plasmas. Additionally, we must at present judge the use of optical diagnostics as *highly risky* in such an environment. This uncertainty alone necessitates a serious rethinking of how to maintain device integrity, not to mention device control, since these functions for present devices and for ITER depend heavily upon optical diagnostics.

These measurement challenges must be met in spite of the fact that diagnostic access to the plasma is being reduced by the need for a high breeding ratio and low streaming loss. In other words, it is almost certain that a DEMO will operate with reduced access, with a reduced set of diagnostics, and with a more self-organized, burning plasma (i.e., one with a reduced sensitivity to external controls), as compared to ITER and present devices. The steady-state nature of the mission brings with it additional new challenges for real-time measurement acquisition, interpretation and analysis, redundancy for crucial measurements, and calibration maintenance.

The following general sub-issues for measurements were explicitly called out in the FESAC Panel report[1]:

- Measurement Capability — Are the possible measurements adequate to provide necessary predictive or control capability?
- Measurement Compatibility — Can the measurements be made in the environment of the device?
- Reliability and Calibration — Will the measurements remain trustworthy for the appropriate time period?
- Interpretation and Analysis — Can the accurate measurements be provided in real time?

🟩 mostly OK 🟨 some R&D needed 🟥 significant R&D needed	measurement readiness	control capable	compatibility	reliability	real-time interpretation
Sufficient for present-day tokamaks	🟨🟩	🟩	🟩	🟨	🟨🟩
Should be sufficient for ITER	🟥	🟨	🟥	🟥	🟨
Should be sufficient for a DEMO	🟥	🟥	🟥	🟥	🟥

Table 1. Assessment of Measurement Research Needs. Note that the "Measurement Capability" sub-issue has been split here into the "measurement readiness" and "control capable" columns. Elaboration of the assessments of the "measurement readiness" and "real-time interpretation" for present-day tokamaks is necessary. Present-day tokamaks have excellent diagnostic capability. However, additional research and development on current devices is needed in those categories to gain crucial understanding in specific areas and to perform crucial validation experiments in support of future needs (see text). Hence the "striped" assessment for those categories.

To highlight the research needs of measurements and diagnostics for burning plasmas we assessed their urgency and magnitude. These are summarized in Table 1, which also illustrates the magnitude of the steps to be taken when moving from present-day diagnostic capabilities, through ITER, to a DEMO device.

In organizing the research opportunities and requirements for making the necessary measurements on steady-state burning plasmas, we have condensed the measurement sub-issues into the following three areas (just as was done when considering these issues within Theme 1):

- Development of the **Measurement Capabilities** needed for physics understanding of issues crucial for the achievement of burning plasmas.
- Development of the necessary **Compatibility of the Measurements, Calibration Strategies, and Reliability Requirements** with the expected difficult environment.
- Development of the necessary degree of real-time **Interpretation and Analysis** of the measurements to facilitate the needed level of plasma control.

Research Opportunity — Development of the measurement capabilities needed for physics understanding of issues crucial for the achievement of burning plasmas. This research opportunity is covered in Chapter 1 and will not be repeated here.

Research Opportunity — Development of the necessary compatibility of the measurements, calibration strategies, and reliability requirements with the expected difficult environment. This opportunity is the most challenging. As has been pointed out, the near-plasma environment will be much more extreme than that of ITER, and, while the ITER *program* is crucial for proceeding to DEMO, satisfying ITER needs will *not* solve longer-term DEMO concerns.

The primary role for diagnostics in a DEMO is device safety, plasma control for operation, and wall protection. Robust control with compatible sensors is a prerequisite. We anticipate that DEMO operational scenarios will be relatively invariant, perhaps reducing the ranges of parameters for diagnostic coverage. The research opportunities in this area are:

- *Research to define minimum set of measurement requirements for DEMO control and safety* — While this will depend somewhat on the operation scenario, an approximate set of measurement requirements can be generated prior to the specification of the scenario. This will guide needed research for making specific measurements. In some cases, the need for research to develop new measurement capability is clear already. In other words, there are some measurements for which we already know that present techniques will not transfer to a DEMO. Thus, we also propose the efforts in the next bullet.
- *Systematic research and development for those measurements for which present techniques are not transferable to the steady-state high-performance burning plasma environment* — Examples of these measurement areas include steady-state magnetics measurements, light extractors, first-wall and blanket instrumentation, monitors for internal components, and first-wall erosion monitors. Keep in mind that the lifetime estimate for conventional

proximity-coil-based magnetics is ~13 weeks, and that the successful use of optical first mirrors must at present be judged improbable.

- *R&D in engineering instrumentation for monitoring in severe nuclear environment, for detectors, and for window materials* — With regard to engineering instrumentation, monitoring must include (at a minimum) confirmation that the first wall and blanket are functioning as desired. With the probable loss of optical monitoring capability, this diagnostic development effort might include work for deployment of "smart tiles," robust instrumentation for plasma facing components (PFCs), blanket modules, and in-vessel components. This research requirement emphasizes the need for a critical set of capabilities in fusion nuclear *technology* that needs to be in place to proceed further. Progress in this area will occur through a well-integrated program of computational models and well-instrumented benchmark experiments.

- *Research to address problems of sensor proximity* — How close must a sensor be to fulfill the measurement requirement? This research need is closely coupled to the research outlined in the next bullet.

- *Research to develop creative, robust diagnostic techniques driven by the need for robust measurement systems and low-risk diagnostic-plasma interfaces* — Novel techniques or approaches that have a high probability for extrapolation to a Component Test Facility (CTF) or DEMO would be fostered in this area.

- *Research and development for in situ calibration techniques for probable measurement systems.*

- Research and development for calibration techniques that can be performed during plasma (deuterium-tritium [D-T] burn) operations.

Research Opportunity – Development of the necessary degree of real-time interpretation and analysis of measurements. This research opportunity is covered in the "diagnosing a self-heated plasma" section of Theme 1. Additionally we note that the control sensor signals and measurement analyses for the control system must be processed in real time. The control sensors will have to be extremely reliable and long-lived. Furthermore, as discussed later in this Theme, the acceptable frequency for transient events is reduced considerably for DEMO-like plasmas as compared to ITER, with greater emphasis on reliability and complete suppression.

Time scales for research and observations concerning facilities and testing. Because of the long lead times for solutions to these challenges, we believe work on them should begin as soon as possible. It is also important to point out that in burning plasma devices and in facilities that would serve as test beds for burning plasma diagnostics, planning for measurements must be an integral "zeroth–order" part of the device *design*. Diagnostics cannot be approached as "add-ons" in these devices. An adequate facility for testing environmental effects, steady-state, reliability, and calibration issues is important for a number of research elements listed here. Access to facilities and run-time for testing are needed in order:

- To test new diagnostic techniques.

- To mitigate environmental effects.

- To establish reliable steady-state diagnostic operations.
- To address calibration issues.

Experience with measurement systems on ITER will be extremely valuable, but incomplete.

INTEGRATION OF HIGH-PERFORMANCE STEADY-STATE BURNING PLASMAS: RESEARCH REQUIREMENTS

Overall Goal: *Create and conduct research, on a routine basis, of high-performance core, edge and SOL plasmas in steady state with the combined performance characteristics required for DEMO.*

The elements identified in the Priorities, Gaps and Opportunities Report as requiring integration to meet this goal were the high-performance burning plasma core, the edge and scrape-off layer plasmas, sustainment of the magnetic configuration and plasma, and optimization of the plasma configuration.

The ReNeW integration panel has identified the core plasma dynamics and the coupling of the plasma edge to the core in a self-consistent high-performance fusion plasma regime as areas for focusing scientific exploration and development. Two supporting elements have emerged as important for achieving these integrated regimes experimentally and providing confidence that the high fusion power regime can be reached: developing high confidence predictive theory and simulation, and a substantial focus on plasma material interactions and material evolution under plasma and neutron loads.

The demonstration of a high-performance plasma core suitable for fusion power generation is a significant integration step in itself. The core plasma issues associated with reaching these parameters are generally broken up along topical plasma physics areas. Key issues and requirements for each are discussed, and must be integrated.

Self-consistent transport and current profiles in alpha-dominated plasmas

*The plasma transport of energy, particles, momentum, and current become strongly interdependent as the core plasma reaches large bootstrap current fraction, high beta, and high ratio of alpha power to input power. **Under these conditions, what are the plasma configurations that emerge from these self-consistent internal physics processes?***

The fusion reactions will generate high-energy alpha particles, which heat the thermal ions and electrons in the plasma as they slow down. The magnitude and profile of this heating in the plasma is determined by the density and temperature of the ions. The profiles of the temperature and density of the ions and electrons are determined both by the heating source and the transport processes for energy, particles, and momentum. The transport is driven by turbulence, which is dependent on the gradients of temperature and density, as well as the profiles of magnetic and electric fields. The plasma generates its own "bootstrap" current that will strongly influence the profile of the magnetic field; it is determined by the profiles of the temperature and density of ions and electrons. As the high fusion power regime is approached, the plasma's own processes will dominate the external sources of heating and current drive, and the plasma will self-organize to a state consistent with its underlying physics. The power from alpha particles (P_{alpha}), in the DEMO regime, will be larger than the externally injected power (P_{input}) by a factor of 4-9, i.e., the ratio $Q=P_{fus}/P_{input} = 20-45$. In addition, the ratio of plasma self-driven current ($I_{bootstrap}$) to the total plasma current (I_{plasma}) is expected to reach values of 0.65-0.90, or perhaps even higher. Although we have concentrated on the plasma transport and current drive couplings, shown schematically in Figure 3, it is recognized that there are also couplings in the MHD area, among fast alpha particles, global instabilities, and the pedestal.

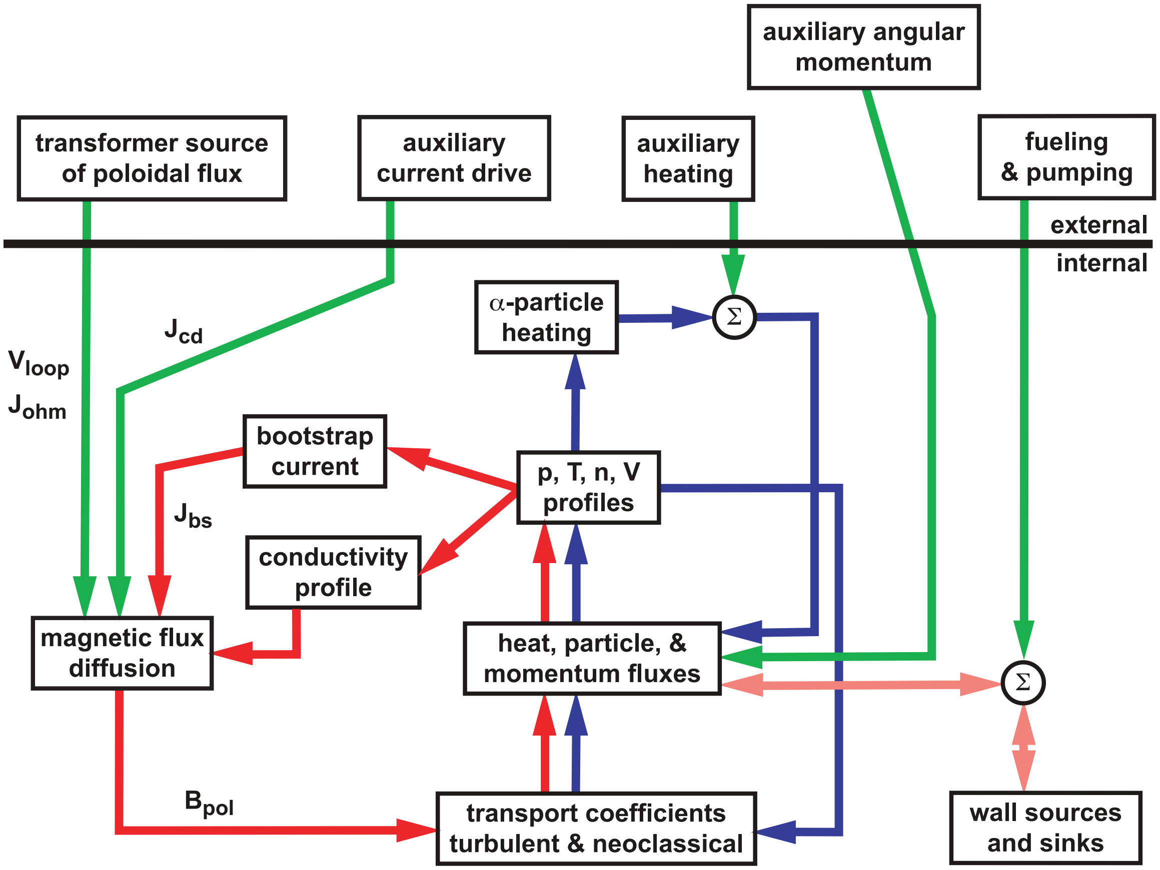


Figure 3: Some of the many feedback loops important in plasma scenarios with strong self-heating and self-driven currents. From P. Politzer et al, Nucl. Fusion 45 (2005) 417-424.

This regime will generate dominant heating of the electron channel, where our understanding is limited, as opposed to the ion channel, where experimental evidence and theoretical understanding have made significant progress. Operation at high density, near the empirical Greenwald density limit, which is traditionally avoided, will become the norm as the requirement to generate fusion power forces the plasma density higher than in deuterium (D-D) experiments. There is limited understanding of the processes determining this empirical limit. The impact of driven and intrinsic rotation on transport and stability, which is only now being explored in detail on existing tokamaks, is difficult to project to future devices, particularly with a dominant alpha heating source. It is recognized that the plasma rotation present on existing tokamaks has a fundamental influence on plasma performance, and an improved understanding of this area will be necessary to optimize the configuration. The current profile and the transport of particles and energy are strongly coupled, particularly when the current and heating source are dominantly from the plasma itself. The longest time scale for core plasma physics is the current profile redistribution time (τ_J). The transport of the energetic alpha-particles and impurities (intentional and unintentional) must be understood to control the distribution of power onto the various material surfaces in the tokamak. All of these issues contribute to the critical need for experimental demonstrations of a high-performance plasma core with the many simultaneous processes associated with a burning plasma as it approaches the self-organized regime.

Magnetohydrodynamic stability

The magnetohydrodynamic (MHD) stability properties of a high-performance core plasma will determine the maximum fusion power density achievable. The understanding of stabilizing a plasma above the no-wall beta limit, as a function of the plasma rotation, background error fields, and feedback control, is critical to accessing the highest possible performance ***. In the burning plasma state with low rotation, fast alpha particles, and strong pedestals, what maximum stability properties will the plasma access?***

Experimental results indicate that in the presence of a conducting wall, even slow plasma rotation – less than 0.5% of the Alfvén wave velocity – is sufficient to maintain plasma stability up to β_N (normalized plasma pressure) values close to the maximum with-wall limit. Although this is encouraging for future large devices where the plasma rotation is expected to be small, a reliable extrapolation to DEMO requires a deeper understanding of the stabilization physics. It is expected that the remaining error fields in the plasma are affecting the stability threshold. Although projections to high fusion power plasmas consider feedback control as the sole stabilization method, it should be recognized that sustained, direct feedback-stabilization without the stabilizing effect of rotation or fast particles has not been demonstrated yet in a high-beta tokamak. The range of plasma betas, in the DEMO regime, is between the value requiring no conducting wall ($\beta_N^{\text{no wall}}$) and that requiring a conducting wall ($\beta_N^{\text{with wall}}$). On a time scale longer than a resistive time, feedback is complicated by the nonideal response of the plasma. The benefits of operating above the no-wall beta limit are clear, and reliable stabilization methods to access this regime are necessary.

The self-heating of the plasma comes from the 3.5 MeV alpha particles slowing down and heating the ions and electrons. Fast ions can drive instabilities, causing enhanced transport of the energetic particles necessary for heating, and possibly damaging the first wall through losses from the plasma. These fast particles can play an important role in the global stability of the plasma, contributing to the access to high beta.

The region near the plasma edge, featuring a transport barrier referred to as the pedestal, strongly determines the core plasma properties. High pressure at the top of the pedestal can support high overall performance in a fusion plasma. The magnitude of this pressure is a key parameter requiring more accurate prediction for DEMO. A key additional issue for the pedestal is that its density and temperature must be consistent with the high-performance core plasma, fueling, and the divertor where particles and power are received.

The barrier gradient is generally limited by MHD stability, and its transient collapses, referred to as ELMs, will not be tolerable in future devices as the associated burst of energy and particles expelled from the plasma severely limit the lifetime of material surfaces. Similarly, other large transients can result in loss of the plasma configuration, and cannot be tolerated in future devices. It is clear that the MHD issues for a high fusion power core can interact strongly, requiring a consistent demonstration in a burning plasma.

Heating and current drive

What heating and current drive sources will be the most effective in creating and sustaining the desired plasma transport and MHD stability properties? As the fraction of plasma current and heating power to the plasma diminishes, which heating and current sources provide new and more effective control tools?

Neutral beam (NB) injection has dominated most of the tokamak experiments around the world, as a reliable and effective plasma heating, current drive, and rotation drive tool, with particle energies less than 120 keV. To provide central deposition in burning plasmas, the particle energy must increase, requiring a transition from the positive to the negative ion acceleration methods, going to 1 MeV particle energy for ITER, and over 2 MeV particle energy for a power plant. The NBs at these energies may drive fast particle instabilities. As one approaches DEMO, these sources must develop solutions to acceleration, cryopump regeneration, and the neutron streaming problem with the large apertures required for NBs.

Although wave propagation and absorption are theoretically well developed in all frequency regimes, to apply radiofrequency power as a tool for control of the pressure and current profiles requires understanding how radiofrequency waves interact in a burning plasma environment. The interaction of ICRF waves and LHRF waves with energetic particles is not well understood, and the coupling of ICRF and LHRF waves from the launchers to the core plasma, over long distances (10-15 cm), must also be quantified, accounting for the exposure of the launchers to the harsh nuclear environment. The interaction of radiofrequency launchers with the edge plasma and ultimately material surfaces must be minimized, to guarantee that power enters the plasma and impurities are not generated. The primary challenge for application of power in the electron cyclotron range of frequencies (ECRF) is technological in nature and related to the need for gyrotron sources at higher frequencies (> 200 GHz), as the magnetic field and plasma density in the plasma increase. An important integration problem for ECRF will be to develop the understanding of the stabilization of sawteeth and neoclassical tearing modes. This requires understanding the coupling of the ECRF to the time-dependent evolution of resistive MHD modes. All radiofrequency sources and NB for driving current noninductively will be challenged by the higher densities typical of burning plasmas, and these sources must be optimized to simultaneously provide current drive, heating, and possibly flow drive, as efficiently as possible, in a plasma that is strongly driven by its internal physics.

Scenario optimization

The experimental goal of research in the area of core dynamics in a D-T plasma is to determine if we can attain the plasma parameters required for a fusion power plant. ***What is the most attractive core burning plasma regime that can be achieved?***

The external control of a high-performance plasma core will become progressively more difficult, relying on the self-organization of the plasma. As the ratio of self-heating to applied heating P_{alpha}/P_{input} becomes larger, the external heating sources become a weaker contribution, tending to be only one-fifth to one-tenth of the total power into the plasma. As the plasma density increases, as it must to generate large fusion powers, the current drive from external sources becomes a smaller fraction of the total plasma current, also about one-fifth to one-tenth , while the self-generated bootstrap current dominates. There is expected to be a net loss of diagnostics compared to those typically found on experimental tokamaks today, as the neutron fluence and power levels increase. A strong reliance on simulations to replace critical measurements is expected to emerge. Multi-level feedback systems would be required on a level beyond that on present tokamaks or even ITER to control the many coupled parameters. Understanding how high a core plasma performance is consistent with a stable and sustainable plasma configuration is a major goal of integrated core plasma studies.

Role of predictive capability in extrapolation to DEMO

Demonstrating plasma parameters at the same level as a fusion power plant (DEMO) is not possible without building the power plant. Therefore, parameters will be demonstrated only up to some level, likely not at precisely the same as a power plant, and often in isolation and not all simultaneously. ***What DEMO parameters can we demonstrate dimensionally or non-dimensionally, and what sets can be demonstrated simultaneously in a given D-T burning experiment?***

This aspect leads directly to the importance of predictive theory and simulation, to bridge the gap from the final pre-DEMO device to DEMO. Here we emphasize the closing window of opportunity to establish a sufficient level of predictive capability. To validate theory and simulation on experiments, advanced diagnostic techniques are required, such as turbulence fluctuation diagnostics, to provide a detailed measurement to compare to a simulation. As the environment in a D-T tokamak becomes more severe from high neutron and plasma loads, many of these diagnostics cannot be utilized. As the most sophisticated diagnostics disappear, the ability to make the most direct comparisons with theory and simulations will also diminish. The full use of D-D tokamaks (including the long pulse Asian devices) and the low fluence ITER D-T tokamak will be critical to establishing a simulation capability that is required to make the step to DEMO. This is discussed further in this Theme.

It is expected that the size of the gaps in demonstrating DEMO-level plasma parameters experimentally, and the confidence in validated simulation for extrapolating across such gaps, will vary, requiring careful planning for proposed devices and physics theory and simulation development. ***How do we plan predictive simulation developments and experimental developments to simultaneously minimize projections to DEMO?***

Interaction with plasma boundary and material interfaces

Present tokamaks have clearly shown the importance of the plasma edge, the scrape-off layer (region between the plasma's magnetic boundary and solid walls) and plasma-material interfaces to the core plasma performance. Considerable effort is spent on "conditioning" the plasma-material interfaces for the highest performance discharges. ***How will self-heated plasmas interact with their material interfaces, and what is the self-consistent core/scrape-off layer/divertor plasma state?***

The time scale for the plasma facing components to come into equilibrium with respect to many processes is not clear. Yet it is an important issue for the viability of fusion power production, where the device would need to operate continuously for periods of about one year. The generation of debris from solid materials in the vacuum region, such as dust, must be kept to levels that will not affect the steady operation of the plasma. Since transients, ranging from ELMs to disruptions, aggravate all plasma facing component issues, they must be reduced to tolerable levels as the high fusion power regime is reached. The core-edge coupling is broken into three primary areas, i) heat loads, ii) particle transport, and iii) material evolution. The self-consistent core and edge plasma demonstration in the high fusion power regime would require better understanding of the following issues.

The heat load issue concerns the consistency of the high-performance core and pedestal with divertor and first-wall power handling. The plasma density and temperature in the divertor are coupled through the scrape-off layer to the edge of the core plasma, and ultimately to the core plasma. The conventional approach to a divertor plasma region that radiates a large fraction of power requires high plasma density, and the compatibility of this with a high-performance core plasma is unclear. ***Finding solutions to the core, pedestal, and divertor will require the understanding of significantly different plasma physics regimes, ranging from the hot, low collisionality core plasma, to the open magnetic field line dominated scrape-off layer, and the high plasma and high neutral density divertor.*** The power conducted through the plasma boundary to the scrape-off layer and finally to the divertor is concentrated in a narrow layer near the plasma boundary. Our understanding of this layer is quite limited, but of critical importance to finding solutions for the survival of divertors at high fusion power. Transients like edge localized modes create pulses of high power, which are intolerable at high fusion power. The balance of power received by various material surfaces is strongly influenced by radiation, with the core plasma radiating some fraction of its power ($P_{rad,core}$), and the remaining power transported to the divertor, where an additional fraction is radiated ($P_{rad,div}$). The tolerable levels of radiation from these regions must be consistent with the underlying physics in these regions.

Particle transport issues that relate to the core-edge coupling include the removal of helium ash from the core plasma through the pumped divertor, the cycling of impurities (both intentional and unintentional) and fuel between the core plasma and plasma material interfaces, particle retention in the solid materials, and the particle behavior in the presence of high power transfer to the solid materials. The solid materials are expected to operate at much higher temperatures than in present tokamak experiments, which may significantly change the retention of fuel in these materials. The physics of the high-density limit, when the plasma density reaches the empirical Greenwald density limit, is only partially understood. The fundamental limit is believed not to be an average density, but to be driven by processes in the scrape-off layer. Since plasmas in the high fusion power regime are likely to be operating near this density, the underlying processes will need better understanding. The purity of the core plasma, measured as an effective charge (Z_{eff}), is directly related to the externally supplied fueling and pumping, the particles that are recycled and eroded from the solid material surfaces, and impurity particle transport.

The evolution of the solid materials interfacing plasma under both neutron and plasma loads is considered one of the most serious issues for the high fusion power regime. ***The materials are affected by possible melting, fast ion or electron impingement, neutron damage, tritium and deuterium retention, and erosion and redeposition by plasma. The impact on the core plasma arises primarily from the generation of impurities, dust, and larger material removal such as bubbles and flakes.*** As the high fusion power regime is approached, these effects become significantly aggravated. Although material lifetime is determined by these processes, a material's behavior can adversely affect the core plasma long before its properties dictate replacement.

Potential "game changers"

What new areas can provide innovative solutions to some of the more uncertain issues in the high-performance fusion plasma regime?

Although there are conventional solutions to a number of issues for the high-performance steady-state burning plasma regime, some of these have significant uncertainty, requiring innovation for their resolution. Consideration should be given to the use of three-dimensional magnetic fields, based on stellarator research, to influence MHD and avoid disruptions. In addition, these fields may provide rotational transform, reducing the current drive requirements, and may also provide resilience to the density limit observed in tokamaks. The exploration of advanced divertors, including liquid metal approaches, should be pursued to understand their potential for handling the high particle and power loads in a fusion power plant as well as the particle control and material evolution issues. The possibility of new, more powerful controls, with flow shear or alpha particle "engineering," on internal plasma profiles may be possible because of the strong coupling in the high-performance fusion regime; it should be explored.

Summary of Key Integration Goals

While various aspects of high-performance steady-state burning plasmas have been discussed in this section, the greatest challenge lies in integrating all of the physics and target parameters simultaneously. Some of these are summarized in Table 2, which compares key global and boundary-related parameters expected in ITER steady-state scenarios with those in two potential DEMO designs. Large gaps in a number of areas, including bootstrap fraction, alpha heating fraction, duration, and heat and neutron flux, can be noted. These motivate new experiments as discussed in the Research Thrusts.

	ITER-AT	ARIES-I	ARIES-AT
β_N (%)	3.0	3.2	5.4
$I_{bootstrap}/I_{plasma}$	0.48-0.68	0.68	0.89
n/n_{Gr}	1.0	1.04	0.95
q_{cyl}, q_{95}	3.8-4.5	4.4	3.0
Z_{eff}	1.4-2.0	1.73	1.83
P_{rad}^{core}/P_{input}	0.2-0.3	0.48	0.36
P_{rad}^{div}/P_{input}			0.43
$(P_{alpha} + P_{input} - P_{rad}^{core})/A_p$, MW/m^2	0.14	0.45	0.56
$(P_{alpha} + P_{input} - P_{rad}^{core})/P_{input}$	1.36	2.52	6.25
P_{alpha}/P_{input}	1	3.8	8.8
$<N_W>$, MW/m^2	0.6	2.5	3.3
τ_J, s	200-400	300	275
Duration, s	3000	~3x10^7	~3x10^7
H diffusion depth (m)	4x10^{-5}		0.61
Wall temperature, °K	400		700-1300

Table 2. Key Parameters for ITER and Reactor Designs.

VALIDATED THEORY AND PREDICTIVE MODELING: RESEARCH REQUIREMENTS

Overall Goal: Through developments in theory and modeling, and careful comparison with experiments, develop a set of computational models that are capable of predicting all important plasma behavior in the regimes and geometries relevant for practical fusion energy.

Two general areas of importance for meeting this FESAC goal were identified by the ReNeW panel.

Gap: In the past, there has been no separate funding effort for predictive modeling and validation within the US fusion energy sciences program. Instead, such efforts have been supported as a portion of larger experimental or model development activities. The panel believes that successful development of a validated predictive capability will require a more dedicated and focused effort than currently exists within the program.

Gap: The panel recognized that the integration of a high-performance steady-state plasma with a DEMO-relevant wall/divertor solution and the demonstration of fully steady-state disruption-free operations will not be possible in ITER, but is nonetheless essential for proceeding beyond ITER. Thus, in addition to being of use in planning and executing experiments on ITER, predictive modeling capabilities are needed to investigate and resolve integration issues for a DEMO reactor.

In the discussion below, key research opportunities and requirements identified by the panel are described according to topical science area. It should be noted, though, that many of the opportunities lie in the integration of multiple topical areas.

Core Plasma Physics (including Transport and MHD Stability)

Science Opportunities:

- How do we tailor the pressure and current profiles to achieve optimized steady-state burning plasma tokamak operation in which self-heating dominates?
- How can we produce high-performance, quiescent, steady-state plasmas?
- What physics governs transport of fusion fuel and reaction products as well as wall-generated impurities? What causes core transport barrier formation?
- How rapidly will large reactor plasmas rotate, and what effect does rotation have on performance limits?

Research Requirements

Progress here requires advances in theory, modeling, and experiment. An analytic understanding of underlying first-principles physics models is required for verification of models. Advances in treating multi-scaled or multi-physics problems are needed. For example, improved coupling of transport-MHD and RF-fast particle physical processes for predictive profile evolution and steady-state operations are needed. Systematic development of such a capability should be planned, in collaboration with analytic theory, development of reduced integrated modeling, and dedicated experimental validation efforts that include synthetic diagnostic development.

Edge/Scrape-off Layer (SOL)/Divertor Plasma Physics

Science Opportunities:

- What are the power and particle loads on divertor plates and first wall in a high-performance plasma?
- How will a fully steady-state reactor-relevant wall affect the core plasma performance, and what approaches can be used to control the erosion, transport, and redeposition of wall material?
- How do we adequately fuel the plasma and remove fusion reaction byproducts?
- How do we manage and control the tritium inventory in a steady-state reactor?

Research Requirements

Significant advances in theory, modeling, and experiment are needed to address the science opportunities listed above. Particularly challenging, given large fluctuation amplitudes and a lack of clear scale separation, are the calculation of self-consistent electric fields and self-generated flows, and the proper treatment of open field lines. In addition, improved models for sheath physics, including radiofrequency and kinetic effects and SOL/first-wall/antenna geometry, and physics-based models for plasma-wall interaction, including multi-scale material physics, chemistry, and morphology, are needed. New measurement requirements include profile, flow and fluctuation measurements with better geometric coverage; measurements of Ly-α and other radiation sources in dense divertor plasmas to account for radiation transfer and opacity effects; and neutral density in the divertor, SOL and edge pedestal to understand fueling and divertor performance. Measurements of radiofrequency sheaths and heat flux measurements on plasma facing components using infrared thermography and calorimetry with extensive 3-D coverage, and in-situ erosion and redeposition measurements, are needed for rigorous testing of thermal load predictions as well as prediction of impurity generation rate and distribution. Data from integrated high heat flux, steady-state confinement devices with relevant wall conditions, complemented by basic plasma physics devices and plasma-wall interaction test stands, will be required for validation.

Energetic Particle Physics

Science Opportunities:

- Will the formation of energetic particles in burning plasmas lead to the generation of new instabilities? If so, what types of instabilities? What are their thresholds for excitation? How do they grow and saturate?
- Do these instabilities lead to a state of turbulence and in what way might this turbulence couple to the thermal pressure gradient-driven turbulence?
- How do the energetic particles and/or the background plasma turbulent transport, MHD equilibrium and RF driven currents respond to energetic particle modes and/or turbulence?

Research Requirements

Theory, computational, and experimental work is needed to understand the nonlinear saturation and development of a new regime of turbulence and associated transport due to fast-particle physics. New methods for long-term simulations are required that can track the evolution of fast particle generation (which occurs on confinement time scales) while simultaneously capturing the short time scale fast particle/wave interactions driving the turbulence. Emerging diagnostic techniques (e.g., energetic ion distribution and loss diagnostics) should be incorporated into existing laboratory-scale and confinement experiments, and coupled to direct measurements of Alfvén Eigenmodes (AE) and Alfvén turbulence using, for example, reflectometry, beam emission spectroscopy (BES), correlation electron cyclotron emission and other emerging spatiotemporally resolved diagnostics that have traditionally been used to study drift turbulence and transport. Development of local magnetic field fluctuation measurements for determining amplitude and mode structure would also be valuable. Synthetic diagnostic development in models is also needed.

Disruptions: Prediction, Avoidance, and Mitigation

Science Opportunities:

- Can the probability of a disruption event be reduced to something less than one disruption per year while operating a steady-state advanced tokamak at high performance?
- Can we successfully and accurately predict the approach to a disruption event and initiate control techniques to avoid the disruption?

- If a disruption becomes unavoidable, can we then predict the onset, evolution and final termination of disrupting discharges due to all possible instability mechanisms (e.g., density limit radiation collapse, vertical displacement event, beta-limit, etc.) in a DEMO device and use this capability to design a mitigation scheme that will perform with high confidence?

Research Requirements

The fundamental question of disruption probability can only be definitively answered by a new generation of very long pulse (likely many days long) discharges coupled with extensive operational experience of such devices. No such experiment exists today. Integrated modeling of disruptions, which would feature nonlinear MHD simulations for realistic tokamak geometry, including core, scrape-off layer, plasma facing components, and eventually even coil structures to calculate halo currents, must occur. Such a capability could first be developed and experimentally validated for ELMs, then resistive wall modes (RWMs), and finally disruption events. Modeling must eventually successfully predict performance of disruption mitigation techniques with sufficient confidence to make design selection for DEMO. Reduced physics disruption models must be integrated into a real-time control system for disruption prediction and detection, avoidance, and mitigation. Improved diagnostics of disruptions, including runaway electron generation and transport, as well as development and study of potential mitigation techniques, must be performed in existing experiments. Validation experiments using emerging models of unmitigated and mitigated disruption events can provide predictive capabilities for ITER and then DEMO.

Pedestal and ELM

Science Opportunities:

- Can we accurately identify the physics trigger(s) for formation of the edge transport barrier, and translate this into a physics-based "L-H mode" transition threshold prediction?

- Can these predictions then be extended to fully saturated, steady-state wall conditions? Can we predict the height of the pedestal with a physics-based understanding? Can we accurately predict ELM onset, dynamics and the associated heat and particle load on PFCs?

- Can we understand the behavior of ELM control and mitigation schemes sufficiently well to confidently predict their performance and impact upon the pedestal conditions that will occur in a DEMO?

- Can we predict the coupling of radiofrequency, beam and pellet sources across the pedestal?

Research Requirements

A successful predictive pedestal model is inherently a multi-scale problem involving the long-time evolution of a turbulent system coupled with the evolution of the plasma profiles. New approaches are needed if a first-principles model of this system is to be created. Validation will require new measurements (e.g., neutral fueling of the pedestal, field penetration from ELM control coils) as well as significantly improved multipoint measurements of turbulent density, potential, magnetic field and temperature fluctuations and profile. Pedestal gradients may invalidate the ordering assumptions of gyrokinetic simulations, and may therefore force consideration of full kinetic simulations, at least under limited conditions, to test if and when the gyrokinetic models break down. Comparison of experiment and simulation will likely require development of new synthetic diagnostics (e.g., BES, probes, HIBP, and/or GPI) and integration of these tools with edge gyrokinetic simulations for direct experiment-model validation tests. New measurements of magnetic fluctuations and edge current on existing tokamaks would help shed light on the physics of ELMs and ELM mitigation techniques. Many of these studies can be done in existing tokamaks and laboratory experiments. However, it must be recognized that experiments in ITER are required to simultaneously access the collisionality, normalized gyroradius, and neutral fueling depth conditions expected in DEMO.

Radiofrequency Modeling for Heating and Current Drive

Science Opportunities:

- How do alpha particles affect radiofrequency wave propagation and absorption?
- What possible transport effects from radiofrequency waves on alphas may be expected?
- What degree of plasma flow will be generated by radiofrequency waves in ITER and DEMO?
- What are the current drive and heating power requirements and degree of location control in ITER and DEMO?

Research Requirements

The validity of the geometrical optics wave propagation approach in H-mode with high pedestal densities and steep gradients must be examined and perhaps replaced with full wave approaches. Improvements in wave particle interaction modeling, particularly for electron cyclotron wave based approaches, are needed. Accurate modeling of the edge geometry, including the PFC-RF sheath interaction and edge gradients that are on the scale of the radiofrequency wavelength, is needed. These tools must succeed in predicting radiofrequency coupling and loading of an antenna to the edge plasma in the reactor environment, where antenna-plasma separation will be larger than in present-day experiments. Integrated modeling needs include development of RF-MHD and RF-turbulence interactions. Diagnostics for launched wave and mode converted wave detection, as well as fast particle detection, should be deployed and compared with synthetic diagnostics implemented in radiofrequency codes. Radiofrequency probe arrays in the SOL would allow improved radio frequency sheath physics and wave coupling understanding.

Integrated Modeling

Science Opportunities:

- How can modeling bridge the disparate spatiotemporal scales that link different aspects of high-performance steady-state burning plasma physics?
- Can the current approach, invoking scale-separation arguments to decompose the problem into a slowly evolving background coupled to a rapidly evolving spatiotemporal

scale, adequately handle the broad range of relevant problems in the integrated modeling area? If not, what alternate approaches can be applied?

- Can the reliability of each component, as well as the framework connecting components, be improved to the point where integrated models can be used to routinely plan and analyze burning plasma experiments in ITER?

Research Requirements

Theory is needed to improve the fundamental techniques used in multi-scaled simulations. Improved reduced models using the parameterization approach are needed for plasma startup and subsequent evolution of a burning plasma discharge toward the final burning plasma configuration, including episodic MHD phenomena, plasma-wall interactions and turbulent transport over a wide range of conditions. A reliable prediction of the conditions needed for the L-H transition is critically needed, particularly under conditions with a saturated reactor-relevant wall, along with a model for the subsequent development of the density, temperature, and velocity pedestals. Models are needed to predict the onset of ELM activity as well as the plasma evolution through complete ELM cycles. It is also important to predict the performance of ELM control techniques, now poorly understood. A neoclassical tearing mode model is needed that includes the effects of mode coupling and rotation. An improved model of the core "sawtooth" crash is needed, which must include the effects of fast ions and predict the fraction of magnetic reconnection. Transport models must be improved to predict transport near magnetic axis and in the high confinement mode (H-mode) pedestal and SOL region, rather than relying upon experimentally determined values at an arbitrary outer boundary. It is likely that 3-D transport models must be developed for regions near helical magnetic islands, and a dynamic model is needed for neutrals recycled from plasma facing components into the SOL and pedestal region. Advances in radiofrequency antenna-edge plasma coupling, wave propagation and absorption, and fast particle effects are also needed. Improved communications between first-principles physics simulations and reduced models are needed to test and modify reduced models as needed during the evolution of a simulated discharge.

CONTROL: ADDITIONAL RESEARCH REQUIREMENTS

Overall Goal: *Investigate and establish schemes for maintaining high-performance burning plasmas at a desired, multivariate operating point with a specified accuracy for long periods, without disruption or other major excursions.*

INTRODUCTION

The control research needs of ITER are discussed in Theme 1. Control research under Theme 2 focuses on the needs of DEMO and an eventual power reactor. Fusion power plants are expected to be among the most complex systems ever to operate for periods of many months without shutdown. Many of the operational control issues are similar to those encountered in present-day fusion experiments and process-controlled plants. The number of regulated subsystems, the range of time scales, the complexity of the highly coupled nonlinear multivariable control problems, the complexity of the technologies, and the nuclear safety issues involved make it unlike any other large-scale process control problem, as well as a significant extension beyond present devices and ITER.

Development of fusion science and technology has been paced (in many cases enabled) by developments in advanced control solutions, usually implemented in operating fusion devices. Present research programs and presently operating devices have already produced general solutions for some of the control problems expected to face ITER. Given sufficient support, they are expected to identify most or all of the remaining needed solutions before ITER operates. Such programs are already exploring some issues expected to occur in DEMO, but it has been long recognized (and specifically identified as a gap in the Priorities, Gaps and Opportunities Report) that present worldwide plans for fusion development do not provide the means to solve many of the control problems in DEMO (or an eventual commercial reactor). While many ideas to address the gaps exist, enhanced research is required.

The research required to establish the basis for control of DEMO will involve expanding the frontier of advanced mathematical algorithms for the unprecedented complexity of the fusion reactor control problem. This core effort necessitates specific forms of physics understanding, often uniquely driven by control requirements. Physics understanding in turn enables development of new models for all relevant plasma and auxiliary systems that are an appropriate balance of accuracy and computational tractability. Breaking the new ground required in this field and reducing fusion power plant control to practice will require decades of effort, highly coupled with the many elements of fusion science and technology development. While present devices and research have begun this effort in limited ways, the revolutionary solutions required for DEMO imply a need to begin an accelerated and systematic research activity soon. Because present and near-term devices are beginning to explore regimes with highly coupled nonlinear complexity, this accelerated effort will have an immediate impact on the experimental fusion program.

GENERAL CONTROL RESEARCH GAPS AND NEEDS FOR DEMO

The sections following describe individual areas with identified research gaps and requirements. However, all of these research areas in fusion plasma control share certain common requirements. Model-based control requires ***validated control-level models*** for design of relevant systems in active control or detection and response loops. Sufficiently ***detailed simulations*** and associated models must be developed to verify both controller performance and implementation prior to use. ***Control algorithm design*** approaches and solutions must be created for each area, in many cases requiring additional mathematics and algorithmic understanding at or beyond present frontiers in the control field. ***Real-time computational solutions*** are required for analysis and interpretation of plasma and plant state (e.g., real-time analysis and prediction of proximity to stability boundaries) as well as implementation of complex control algorithms. In many cases new control ***actuators and diagnostics*** must be developed with the necessary effectiveness, dynamic performance, and accuracy. ***Experimental demonstrations*** of control schemes and specific controllers in relevant environments (sustained duration, neutronics, relevant actuators and diagnostics, operating regimes, etc.) will be essential. The ability to ***quantify reliability*** of control performance for a reactor (including nominal operation, response to faults, and probability of fault scenarios) will be required to meet nuclear and licensing constraints. In many cases ***im-***

proved physics understanding is required, including sufficiently detailed computational physics models and control-level models. Control ***models*** frequently require far less accuracy and/or precision than the goals of detailed physics codes, although measurement accuracy and precision requirements tend to be very high for real-time control. ***Improved mathematics and algorithmic understanding*** is required in most cases for development of the required control schemes and controller designs. The engagement and coordination of cross-disciplinary expertise, including physics, control mathematics, and fusion system engineering, is essential to fill these research gaps.

OPERATING REGIME REGULATION

Regulation of fusion plant system operating regimes generally includes the equilibrium shape and position state, bulk quantities (such as plasma current and beta), various profiles (including current density, pressure, density, and rotation), and the divertor configuration. Particular solutions needed for power plants include operating point regulation for noninductive, true steady-state, self-heated, sustained duration operation, likely with high bootstrap current fraction. Specific research requirements identified for operating regime control include developing algorithms and approaches for use of superconducting coils in reactor regimes and noninductive operation, as well as general methods for integrated regulation of global parameters (e.g., plasma current, stored energy, fusion power output). Plasma shape control schemes must be developed consistent with limited availability of diagnostics and coils. Experimental demonstrations of operating regime control of self-heated plasmas must be performed in discharges with high (up to 90%) bootstrap current fraction.

PLANT STARTUP AND SHUTDOWN

Control demands for plant startup and shutdown are expected to be challenging in DEMO, owing to limitations in central solenoid size and available flux. For example, startup of several ARIES point designs with attractive cost of electricity will require several hours, relying on initiation and burn-through using a small solenoid followed by pure noninductive rampup[2,3]. Research is required to develop and experimentally demonstrate such a plasma current ramp-up to the steady-state regime, followed by ramp-down using minimal inductive current drive.

KINETICS

Outstanding solutions required for kinetic control (e.g., temperature and density profiles) in plant operation include steady-state fueling solutions and divertor kinetic operation (including heat flux, radiation state, impurity and pumping regulation, and control of advanced configurations with high multipole moment magnetic topologies). Research requirements in this area include development of methods for and demonstration of coupled performance in the core and divertor (neutron rate/fusion power, H-mode confinement state, power flow through pedestal into SOL). Divertor operation will require development of methods for regulation of the divertor magnetic configuration consistent with a high nuclear fluence reactor environment, which may be particularly challenging for high magnetic multipole configurations (e.g., “Snowflake” or “Super-X” divertors, discussed in Theme 3) with stringent requirements on divertor coil proximity and diagnostics. Integrated control methods for simultaneous fueling regulation and burn control must also be developed and demonstrated.

FUSION PLANT

Control requirements specific to fusion power plant operation include blanket operation, regulation of output power across a wide range to match the grid load, supervisory control under sustained duration conditions to ensure high availability and high reliability performance, and remote maintenance control solutions. Integrated blanket and plasma control must simultaneously balance fusion power (plant output) requirements with blanket breeding efficiency requirements. In particular, a blanket operation control solution must be developed consistent with power plant requirements (breeding ratio [BR] must be regulated in a narrow region just above 1.0: BR[max]> BR > 1.0, where likely BR[max]~ 1.05-1.1 to respect on-site tritium inventory limits). Methods for fusion output power regulation must be developed and demonstrated with high efficiency, stability, and performance robustness for true sustained duration of many months without downtime.

STABILITY

High-performance power plants are often envisioned to operate in regimes (e.g., advanced tokamak) very close to or beyond passive stability limits. Robust operation in such regimes requires active and reliable stabilization of many instabilities including resistive wall modes, neoclassical tearing modes (NTMs), energetic particle modes, thermal instabilities, axisymmetric instabilities, ELMs, and core “sawteeth.” For example, robustness to NTMs may require suppression of sawteeth as a seed for island formation, as well as preemptive current drive at resonant surfaces, to raise the metastability threshold. Research requirements include developing and demonstrating methods of passive and active RWM stabilization consistent with DEMO operation. Edge localized mode control solutions consistent with DEMO must be designed and demonstrated in relevant steady-state plasmas.

Development and experimental demonstration of control solutions for the highly constrained, multivariable problem of burn regulation will be essential. An ARIES-AT-like DEMO would operate in a nominally stable regime relative to thermal instability, but driven limit cycles or instabilities from the coupled dynamic system must be prevented. The needs, mechanisms and control solutions for fast-particle instability suppression must be determined, consistent with reactor requirements. Neoclassical tearing mode suppression scenarios and control solutions must be developed for DEMO, consistent with reactor conditions and sustained duration. (A key question: is suppression by electron cyclotron current drive achievable and consistent?) Improved understanding of NTM physics, effects of suppression mechanisms (e.g., localized current drive, profile modification, sawtooth stabilization), experimental demonstration, and certification of performance will all be required.

OFF-NORMAL EVENT AND FAULT CONTROL AND RESPONSE

Creating an attractive fusion power plant requires an integrated, comprehensive control solution that provides high confidence operation with quantifiable reliability. A key part of this solution is demonstrably reliable control in proximity to stability boundaries, including excursions expected during nominal operation, as well as off-normal or fault-triggered excursions. Here we define off-normal events as those transient phenomena that lie outside the envelope of noise and disturbances that can be robustly stabilized. Such events may be recoverable (but are not robustly controlled), or may be unrecoverable; they must be responded to in an appropriate way to minimize machine damage and down time. Their probability of occurrence must be reduced to an acceptable level for power plant operation, and this must be accomplished through control design. Reliable operation under nominally disturbed regimes (e.g., intermittent sawteeth, ELMs, or transient magnetic island growth) falls under the stability research category. However, novel algorithmic control and response solutions beyond nominal control are also needed for excursions requiring transient changes of operating regime (e.g., response to loss of a diagnostic channel or key actuator), rapid but controlled shutdown (e.g., identification of an impending unrecoverable state), or emergency uncontrolled shutdown (e.g., identification of immediate unrecoverable state and insufficient time for controlled shutdown, often envisioned as serving to mitigate potential system damage).

Enabling elements of these solutions include a quantifiably reliable systematic approach and corresponding integrated system for response to off-normal events and fault-triggered excursions, real-time predictors for proximity to key operational limits, algorithms and mechanisms for mitigating damage, and recovery and cleanup strategies to rapidly restore plant availability. This will require development and validation of theory-based predictors for proximity to stability boundaries, including assessment of the potential for and development of empirical data-based learning systems. Provable algorithms for response to various off-normal or fault events must be developed to deal with degradation or loss of diagnostic data or of actuator performance, impending or current plasma regime excursion, unexpected disturbances, and impending or current loss of controllability. Control scenarios for controlled (“soft”) shutdown, as well as for preventing or minimizing device damage during large-scale off-normal events, must be developed.

RELIABILITY AND CERTIFICATION

Key solutions are required to quantify performance, risk, and reliability of an operating fusion power plant. The results of this area of research will enable licensing, certification, and economic attractiveness of a realizable power plant. Research requirements identified for control reliability and certification include methods for executing performance assessment (systematically quantifying performance reliability), and methods for failure modes and effects analysis, a systematic approach to identifying potential system failure modes, causes, and effects on fusion plant operation.

MODELING AND DESIGN

Most of the requirements for comprehensive high reliability control of fusion power plants rest on the ability to model and design control solutions with specified and/or quantifiable performance. This in turn requires computational tools for producing control-level models, integrated and sufficiently comprehensive simulations, and real-time predictive models for on-line controller adjustment or operating regime identification. Research requirements identified for control modeling and design include computational tools for producing control-level models and simulations, ideally derived from more complex models, and control-level and intermediate-level simulations. These will enable development and testing of control algorithms and verification of real-time implementations. Models required include all elements of a fusion plant system: relevant plasma responses, actuator responses to commands, diagnostic signal mappings to relevant physics, etc.). Integrated, comprehensive plasma and plant simulations will also be needed, primarily for discovery of emergent phenomena, unexpected interactions, commissioning, performance and reliability quantification, and certification. Real-time predictive models will be needed for on-line operating point identification, controller adjustment and re-design, stability boundary proximity identification and so on. Models and simulations must be validated against relevant experimental data to confirm the required accuracy.

ALGORITHMS AND APPROACHES

The requirement of producing quantifiably high-performance operational solutions requires research to produce novel control algorithms. This research includes mathematical solutions for nominal high-performance control, as well as supervisory and off-normal response algorithms with provable reliability. Control schemes must be developed for all required regulated parameters and stabilized modes, including multi-physics integrated schemes where appropriate. Design tools and solutions must be developed to produce the necessary real-time, model-based control algorithms (mathematical control solutions) with associated gains, point designs, and provable high confidence performance.

As a specific example of the control research needed, a reactor design that operates near stability limits to achieve high performance will require active suppression of tearing modes. Present-day research has demonstrated the need for complex nonlinear control algorithms to locate and align current deposition regions with relevant magnetic islands, to synchronize modulated current with the island phase, and to sustain the alignment and phase synchronization as plasma conditions evolve. The corresponding algorithms used in a reactor must be much more robust than present solutions, and must operate correctly and effectively upon first-time use. At the same time, the relative leverage provided by heating and current drive systems in self-heated burning plasmas will be much smaller. These requirements necessitate a robust design based on high accuracy models: the task is beyond the capability of empirical tuning, and in any case little opportunity will exist in the commissioning process to achieve such tuning. Mathematical control design methods can guarantee a specified level of reliability and robustness, based on the accuracy and reliability of the models on which the design is based, and can squeeze maximum effectiveness from limited actuators. In exactly the same way as envisioned for fusion reactors, model-based design of complex multivariable control algorithms is what enables modern commercial aircraft to fly well and safely in their first flight tests, and indeed in all of their commercial flight operations.

TRANSIENT PLASMA EVENTS: ADDITIONAL RESEARCH REQUIREMENTS

Overall Goal: *Understand the underlying physics and control of high-performance magnetically confined plasmas sufficiently so that "off-normal" plasma operation, which could cause cat-*

astrophic failure of internal components, can be avoided with high reliability and/or develop approaches that allow the devices to tolerate some number or frequency of these events.

INTRODUCTION

In Theme 1, the ITER research requirements associated with large transient or off-normal events, which included disruptions and associated runaway electrons, large ELMs, and bursts of energetic and alpha particles ejected by plasma instabilities, were described in Theme 1. The goal of this section is to extend the research requirements to create predictable high-performance plasmas. While the issues are similar for ITER and DEMO, there are both quantitative and qualitative differences in addressing the broader issue of high-performance plasmas. Quantitatively, as will be discussed, many of the issues identified for ITER will become more difficult assuming the same design approaches are used for DEMO. Qualitatively, DEMO may be able to incorporate different design strategies including the choice of magnetic configurations. These differences result in additional research requirements to assess the potential for different solutions. In this section, the research requirements identified in Theme 1 will not be repeated though they are, in general, applicable to Theme 2.

In considering this, the Priorities, Gaps and Opportunities Report stated:

"Avoiding or mitigating off-normal plasma events in tokamaks is very challenging, particularly in the Advanced Tokamak (AT) performance regime (high Q, high beta, high bootstrap fraction, steady state) anticipated for DEMO. The issue appears to have only two possible plasma-based solutions:

> 1. Discover and develop improved techniques to predict and either avoid or mitigate off-normal events in an AT-regime tokamak with a high degree of confidence. A successful DEMO design must provide for recovery from damage caused by low-probability off-normal events within the availability, safety, and environmental constraints of an economically attractive electric power source.
>
> 2. Improve the understanding and performance of other confinement configurations that either avoid off-normal events or allow more confidence in their control. A successful non-tokamak DEMO design must be based on demonstrated capability of steady-state, high-beta confinement and other properties consistent with providing the Q, availability (including recovery from any off-normal event), safety, and environmental features of an economically attractive electric power source. The stellarator is the most advanced configuration that has the potential to meet these requirements."

DISRUPTIONS

The severity of most consequential effects of disruptions increases for DEMO and beyond. Electromagnetic loadings and area/size-normalized body forces increase modestly (~3-fold) from present tokamaks to ITER and are comparable in DEMO. Thus, while structural loadings associated with the plasma current decay (or motion owed to a vertical displacement event [VDE] with resulting generation of in-vessel halo currents) increase modestly for ITER, the effects can be accommodated in the corresponding structural designs. However, the impact on designs for DEMO, which needs to breed tritium, has not been fully assessed. If currently available first-wall materials were utilized and the same design assumptions used for ITER characterizing disruptions were applied to DEMO, then either the blanket would fail to breed tritium or the availability goals would not be met due to component failure.

The time-weighted energy deposition in ITER and DEMO will be an order of magnitude greater than in present tokamaks, and may approach or exceed thresholds for onset of tungsten surface melting (or carbon vaporization). Even preemptive mitigation will result in uniform-deposition loadings on the first-wall surfaces that approach the respective wall melting thresholds for ITER, and this problem will become greater in DEMO due to its likely higher stored energy and comparable wall area. Avalanche multiplication of runaway electron content may occur during disruption, VDE, or fast-shutdown current decays (the latter effected by gas or pellet injection). The current decay integrated avalanche gain, $G_{aval} \cong \exp[2.5I(\mathrm{MA})]$, is sufficiently high that even very minute levels of "seed" runaway current can convert most of the initial plasma current to runaway electron current. Advanced tokamak modes of operation offer the possibility of increased performance at the same or lower plasma current. Thus, if this problem were successfully solved for ITER, then it would likely be solved for DEMO as well.

Disruption avoidance provides an additional challenge for DEMO relative to ITER. Estimates demonstrate that large improvements in two metrics for "disruptivity" are needed relative to present experiments. (See Figure 11 in Chapter 1.) For ITER, 10-fold reduction in per-pulse disruption rate (discharge setup reliability) and 1000-fold reduction in per-second disruption rate (flattop/burn sustenance reliability) are needed. DEMO requires an additional 100-fold and 1000-fold reduction beyond ITER.

As noted earlier in Theme 1, accurate prediction of an impending disruption is a key element of both avoidance and mitigation of disruptions. This will rely on detailed plasma measurements, which are especially challenging in DEMO, as discussed in the section on Measurements. For future advanced tokamak burning plasmas it will be essential to develop plasma operation and control procedures that ***avoid***, wherever possible, occurrence of disruption onset, to identify means to reliably ***predict*** pending onset of disruption, and to have means available to ***mitigate*** the consequences of disruptions that cannot otherwise be avoided. An additional consideration is that even the techniques used in present machines to avoid disruptions, by modifying the current and pressure profile, are challenging to incorporate in DEMO, as discussed later in this Theme in greater detail. New techniques, which modify the profiles but require little auxiliary power, may have to be developed. In developing strategies for disruption prediction, avoidance, and mitigation, it will also be essential to develop quantitative metrics to ***characterize and model*** disruptions and disruption consequences. Substantial open issues exist in all four categories — prediction, avoidance, mitigation and characterization and modeling — and these gaps are larger for DEMO than for ITER, which are described in Theme 1. The research requirements in this area can be summarized by:

- Determine the level of steady-state performance that tokamaks and STs can achieve and remain disruption free.

- Develop measurement techniques and actuators compatible with the requirements for a tokamak or ST DEMO.

Disruptions in stellarators

Disruptions in stellarators have only been observed during fast current ramps in discharges with ohmically driven current. With the advent of high-power radiofrequency and neutral beam injection (NBI) plasma heating systems, most stellarators have operated with little to no ohmic current during the last several decades. In early experiments in low-shear stellarators, disruptions did not occur if sufficient vacuum rotational transform was applied ($\iota \geq 0.14$). As in tokamaks, disruptions are avoided by keeping $q > 2$ at the boundary, or avoiding fast current ramps. In stellarators, they are also passively avoided by limiting the plasma current and proper design of the vacuum rotational profile. Therefore, rotational transform profile control, which is used to improve confinement in stellarators, is likely to be a desirable element of any research thrust targeted to avoiding disruptions in “hybrid” and finite-bootstrap fraction stellarators.

At present, there is no observation of disruptions in stellarators with some of the rotational transform provided by bootstrap current, even at high β. Further research is needed to:

- Determine the pressure limits in stellarators under fusion conditions.
- Extend stellarator research to higher performance plasmas exploring integrated scenarios to ensure that finite-bootstrap current stellarators continue to passively avoid disruptions.
- Determine the level of vacuum transform, or 3-D shaping, necessary to avoid density and pressure-driven disruptions in tokamaks to understand the boundaries of robust disruption-free operation of toroidal plasma devices.

Stellarator discharge termination by radiative collapse

Abnormal losses of stellarator discharges can take place through radiative collapse due to rapid cooling of the plasma. Such a collapse typically occurs if the density exceeds an operating limit, substantial impurity accumulation takes place, or a sufficiently large metallic flake is suddenly injected into the plasma from the divertor during long-pulse operation. The electron temperature typically collapses on a time scale commensurate with the confinement time. In some instances, the enhanced radiation leads to a termination of the discharge while in others, discharges are observed to partially or fully recover. In tokamak discharges a radiative collapse, for example due to the injection of impurities, typically triggers a disruption and sometimes generates runaway electrons.
The research requirements in this area are to:

- Model and compare with experiment the consequences of a thermal quench of a fusion plasma discharge via radiative collapse.
- In concert with Themes 3 and 5, understand and improve steady-state divertor power handling to avoid the uncontrolled injection of microscopic flakes, and develop methods to counter unpredictable radiative collapses after their onset.

These requirements are applicable to both stellarators and tokamaks.

LARGE ELM HEAT FLUX

The heat flux due to large ELMs is a potentially more serious issue in DEMO than in ITER. For DEMO, the fusion power will increase by a factor of ~5 and the combined alpha and auxiliary heating power ~3-4. Though the size of the reactor remains to be determined, it is likely to be about that of ITER. If the incident energy impulse were limited to ~1MJ, and the fraction of energy lost by ELMs relative to power transported through the last closed flux surface were similar in ITER and DEMO, then the frequency of mitigated ELMs will be 50 to 150Hz. Assuming a 50% availability in DEMO and operation for two years prior to replacement of the divertor targets, this implies that 1.5 x 10^9 to 5 x 10^9 ELMs will take place.

There is significant variability in ELM behavior and the mitigation techniques are not fully reliable. This imposes another consideration. Occasional ELMs of the order of 1.5 MJm^{-2} would have to be limited to ~5000 before 10 mm of erosion takes place if carbon fiber composite (CFC) target plates were used. For tungsten targets, incident heat fluxes between 0.5-1.0 MJm^{-2} are predicted to result in edge melting, while surface melting will occur at higher fluxes. This will not only have an adverse impact on impurity influx, but additional fatigue considerations. An ELM mitigation system for both CFC and tungsten targets has to be highly reliable to avoid damage. This motivates research on techniques that fully suppress ELMs with high reliability. The research needs for ELM suppression, ELM-free operational regimes, and operation with small ELMs identified in Theme 1 are applicable for Theme 2, but with greater emphasis on complete suppression. One additional specific research need is:

- Evaluate whether close fitting coils are viable in DEMO for suppression by means of resonant magnetic perturbations.

Stellarator ELMs

Edge localized modes are often observed in stellarator H-mode discharges as well, but the available data and understanding are more limited at this stage. As in tokamaks, stellarator ELMs are associated with strong pressure gradients at the plasma edge. In W7-AS, the estimated energy loss per ELM is estimated as $\Delta W< 4\%$. Accordingly, the research requirement for controlling ELMs in stellarator configurations is similar to the research requirements for tokamaks described in Theme 1. Nonetheless, the stellarator may offer different scenarios for ELM mitigation and suppression in that the appearance of ELMs depends sensitively on the edge rotational transform. Moreover, an ELM-free high-density H-mode (HDH) regime attained in stellarators exhibits good thermal confinement with low impurity confinement times. The high beta regimes in W7-AS and the Large Helical Device (LHD) did not have ELMs.

The stellarator may also differ from the tokamak due to the apparent absence of a critical temperature gradient determining transport, such as the critical gradient expected for ion-temperature gradient mode (ITG) microturbulence in tokamaks. In high-performance tokamaks the ratio of the central to the edge temperature is three to six, but in stellarators it appears a larger ratio of central to edge temperature can be maintained. The property of ITG turbulence does not appear to be universal. In numerical simulations using the GENE code, the W7-X and NCSX stellarator designs have different properties from tokamaks. With appropriate design, it may be possible to maintain good confinement with a cool plasma edge, which would eliminate ELMs and permit detached divertor operation. The research needs for stellarators include:

- Quantitative understanding of the conditions for ELM onset.
- Develop ELM-free operating regimes with low impurity accumulation.
- Develop an integrated solution to high-performance core parameters and a high-density cold edge.

ALPHA PARTICLES EJECTED BY PLASMA INSTABILITIES

The scientific issues associated with loss of alpha particles by plasma instabilities are similar in ITER and DEMO. The large increase in fluence to the first-wall components prior to planned replacement of first-wall components results in greater concern about loss of alpha particles by plasma instabilities and/or ripple, since in D-T reactions both an alpha particle and a neutron are generated. Thus, the effects such as blistering become even more serious.

An intrinsic advantage of stellarators is that they can operate at higher densities than comparable tokamaks (not constrained by the Greenwald limit). Operation at higher density and lower temperature would reduce the drive for alpha-driven instabilities since the energetic particle density scales as $T^{5/2}$ at fixed thermonuclear power. In contrast with tokamaks, in which noninductive current drive efficiency is reduced at high density, stellarators, in principle, do not require noninductive current drive. Energetic fast ion driven instabilities have been observed in stellarators and would need to be studied further. The list of research tasks identified in Theme 1 would have to be extended to include stellarator magnetic configurations. One specific research need is:

- Understand and control particle loss induced by alpha-particle driven instabilities in stellarators.

TRANSIENT EVENT TOLERANT WALLS

An alternative approach to solid divertor targets is to consider liquid divertor targets in a fusion facility. The potential advantage of liquid divertor targets and more generally liquid first-wall components is that an off-normal transient event, such as a disruption, large ELM, or alpha particle loss, might result in a plasma termination or erosion of the surface but not in damage to first-wall components. Experiments using lithium capillary-pore systems have demonstrated the ability to withstand much larger heat fluxes than either CFC or tungsten targets. Work in this area is just beginning and entails both research and technology development, and would have to be done in concert with Theme 3. Research requirements include:

- Characterize the impact of liquid metals on pedestal structure and stability.

- Assess computationally, on test stands, and in fusion experiments, the interactions between ELMs and liquid metal PFCs.
- Similarly, assess the interactions between liquid metal PFCs and disruptions, runaways, and fast lost-ions.
- Develop techniques to flow liquid in the presence of time varying magnetic fields.
- Address tritium retention issues.
- Assess the compatibility of liquid metal surfaces with high-performance plasmas in small, medium and large experimental facilities.

PLASMA MODIFICATION BY AUXILIARY SYSTEMS: RESEARCH REQUIREMENTS

Overall Goal: *Establish the physics and engineering science of auxiliary systems that can provide power, particles, current and rotation at the appropriate locations in the plasma at the appropriate intensity.*

Key gaps identified by the 2007 FESAC panel included[1]:

Gap: Plasma Heating — Even in high-gain plasma, some level of auxiliary plasma heating may be required for startup, sustainment or instability control. This needs to be achieved precisely and efficiently. New systems and technologies have to be developed or expanded to meet the requirements of DEMO.

Gap: Plasma Current Drive — For steady-state operation the plasma current will have to be produced in a nonpulsed (noninductive) manner, and owing to the low current drive efficiencies of most noninductive means, a high fraction of internally generated current (bootstrap current) is desirable. However, high-performance plasmas with high bootstrap currents are susceptible to instabilities, where tearing modes create zones of zero or low bootstrap current.

Gap: Fueling and Exhaust Control — Operation of DEMO steady state for weeks or months at a time, at high fusion power production, requires that the fuel concentration in the core of the plasma be adjustable and renewable.

Gap: Rotation Control — To optimize plasma confinement in high-beta plasmas, edge rotation improves performance by producing radial velocity shear, which acts to stabilize micro-turbulence and thereby improves plasma confinement.

Presently four heating and current drive actuators, electron cyclotron (ECH, ECCD), neutral beam injection, ion cyclotron (ICRF) and lower hybrid current drive (LHCD) are employed widely and are either already incorporated into the ITER design or seriously being considered. Pellets and gas puffing will be employed for fueling. For DEMO a reduced set of actuators is envisioned. Successful operation on ITER will go a long way to qualifying the set necessary for DEMO. For steady-state AT scenarios, including the Advanced Operation phase of ITER, aimed at DEMO and beyond, the economics would dictate that the external power input to a reactor system be minimized. Thus a tokamak reactor system must operate at high bootstrap current fractions. In such a system some auxiliary power is still needed for the balance of current drive not provided by the bootstrap current, for control, and for plasma burn startup. Key requirements for fueling and for each of the potential heating and current drive actuators are summarized below.

Research Requirements for Fueling

Other than the limited auxiliary power, a flexible fueling system and efficient pumping system are all that remains to control and sustain the fusion burn in advanced scenarios. The fueling and pumping systems must deposit fuel at the required locations, control the density profile to maximize the bootstrap current fraction, remove helium ash, control the divertor operating density, and possibly provide a source of external toroidal momentum input to improve plasma stability limits. Additionally, deep core fueling capability for ITER and beyond could possibly improve fusion burn performance by peaking the density profile. It would also increase the tritium burn-up fraction and thus potentially reduce tritium retention in the vessel. To meet these needs, four research opportunities have been identified to develop satisfactory fueling capability for ITER and a DEMO.

Development of steady-state fueling and pumping technology:
Improve present systems and develop new systems capable of steady-state density profile control and helium ash removal. The primary fueling system for ITER is based on pellet injection. Pellet fueling technology needs to be extended for reliable high-throughput use with tritium. Additionally, the pellet fueling performance needs to be assessed on ITER. Because of the uncertainty of present pellet fueling systems for deep core fueling, new fueling systems capable of deep fueling, such as those based on compact toroid (CT) injection or high-speed pellets, need continued development. These systems also have the potential for localized fueling, which should provide some control of the density profile.

Fueling efficiency and isotope mixture control:
Quantify the increase in tritium burn-up with localized core fueling. Present transport codes need to be benchmarked with results from experiments on large tokamaks to improve particle transport estimates. Simulations should then calculate the tritium burn-up fraction as a function of localized fuel deposition in ITER. Simulations and experimental tests of pellet penetration depth, with velocity and size as a function of the pedestal temperature, are needed. Experimental penetration laws for alternate fueling systems are also needed.

Fueling compatibility with ELMs and MHD:
The compatibility of the fueling system with tolerable-size ELMs and acceptable plasma MHD behavior needs to be established. Experimental validation of ELM pacing with pellets and possibly compact toroid injection are needed, as is an experimental demonstration of the compatibility with neoclassical tearing modes.

Advanced scenario modes:
Since ITER relies on pellet fueling to fuel advanced operating modes, operating scenarios for pellet-fueled AT and hybrid modes need to be developed on present tokamaks. The potential capability of CT-based systems for toroidal momentum injection needs experimental validation.

Research Requirements for Electron Cyclotron Heating and Current Drive

The basic physics of ECH and ECCD in a quiescent axisymmetric plasma are well understood. First-principles models have been incorporated in convenient computer codes and well validated against experiment. Electron cyclotron heating and current drive have been applied successfully to a wide range of devices and experimental objectives, including plasma pre-ionization and start-up, plasma heating, current sustainment, and MHD control. In ITER and DEMO, the characteristic dimensionless parameters are not outside the range already explored in experiment. Remaining research opportunities include:

Detailed effects on plasma performance of heating predominantly the electron fluid:
Like the heating by alpha particles in a burning plasma, ECH (like other radiofrequency techniques) differs from NBI in that it heats electrons instead of dominantly heating ions, and it provides no particle fueling and little or no toroidal momentum input. Heating the electrons dominantly is known to cause changes in the particle, heat, and momentum transport. In a burning plasma such as ITER, most of the heat will come from alpha particles via the electron channel. As discussed in Theme 1, this represents one of the main uncertainties in extrapolating from present results, since most present experiments primarily use ion heating, and electron transport is different, and less well understood, than ion transport. This motivates the desire for experiments with dominant electron heating in advance of ITER. Electron cyclotron heating is an excellent choice for this wave heating because its location and profile can be easily controlled and because it can deliver power without affecting the current profile. Alternatively it can drive highly localized currents, which are effective for modifying the current profile and controlling MHD instabilities like neoclassical tearing modes. This in turn is a strong motivator for near-term developments of ECH capabilities.

Staged development program for gyrotrons and related technology:
For DEMO it is projected that 220 GHz 2 MW cw gyrotrons will be required. A staged development program is required to reach this frequency power level and pulse length. For present-day experiments, gyrotrons with at least 1.5 MW are needed to provide sufficient power for experiments with dominant electron heating given limited port availability. Additionally, gyrotrons will need high efficiency, up to 70%, since minimizing power utilization will be important. Achieving a high fraction of Gaussian beam output is highly desirable for cost-effectiveness in coupling directly to the transmission line. In addition, gyrotrons with frequency tunability would enhance the flexibility of ECH. Slow (overnight) step tunability has already been implemented in the ASDEX-Upgrade ECH system. Development of gyrotrons with several steps in frequency of around 5%, and with high power, very long pulse, and high efficiency, would greatly improve the ability to respond to varying operating points of the scenarios and objectives. Fast frequency tunability in steps of 0.5% in a few seconds may also be possible and practical, and this would provide for a tremendously simplified launching system that would need no moving parts and still be able to follow the shifting location of a neoclassical tearing mode for real-time control. This capability would entail additional development of the matching output units and possibly transmission lines.

Development of a robust ECH launcher:
The launchers designed for ITER are probably unsuitable for DEMO. The last mirror, which faces the plasma, has severe cooling issues. This mirror has a very large heat load from the EC waves plus the plasma radiation and neutron heating. Also, the surface conductivity may degrade over time due to neutron damage and deposition of dust. Coolant must pass to the moveable mirror through flexible tubes that may be subject to leaks caused by neutron-induced brittleness or electrical arcs, and bearings are subject to fatigue. An alternate antenna concept for placing the moveable steering parts very far from the plasma ("remote steering") has been demonstrated, but further development is needed.

Research Requirements for Neutral Beam Injection

Neutral beam injection has been to date the most commonly used heating scheme for fusion experiments. This has principally been through the utilization of positive ion beams. For future large, dense plasmas, the limitation on energy inherent in the neutralization of positive ions makes the development of negative ion neutral beams (NNBI) a necessity. Negative ion neutral beams have been developed on JT60-U and this development will continue for JT-60SA. ITER requires NNBI systems that meet nearly all the requirements for DEMO. Successful implementation of an NNBI on ITER will go a long way toward satisfying the development needs. The principal DEMO beam requirement, which none of these development programs is addressing, is continuous beam operation. In all of these systems, the cryopumps must be regenerated after they have accumulated about half the explosive limit of hydrogen, which corresponds to at most a few thousand seconds of beam time. Because negative ion beam systems presently need all the available surface area for pumping, it is difficult to envision installing sufficient excess pumping to allow isolating and regenerating some pumps while others continue to pump. A lithium jet neutralizer, which has been proposed as an upgrade to the ITER neutral beamlines, would solve this problem by eliminating 75 to 80 % of the gas input into the beamline, while simultaneously increasing the neutralization and electrical efficiency, and greatly reducing heat loads on the accelerator and ion source backplate. This could make regeneration feasible.

Research Requirements for Ion Cyclotron Radiofrequency and Lower Hybrid Current Drive

Ion cyclotron and lower hybrid heating and current drive techniques have many common elements that need to be addressed before they could be confidently applied to DEMO. Both schemes utilize complex launching structures that need to be located close to the plasma edge, thereby exposing these structures to a harsh heat, particle and radiation environment. Both schemes couple radiofrequency energy through a region in the SOL plasma where energy can be lost through various mechanisms. Both have the advantage of high-power steady-state power sources already developed. Key research opportunities for these techniques include:

Wave Propagation and Absorption
Experimental studies, coupled with the development of advanced simulation codes during the past 40 years, have led to an unprecedented understanding of the physics of radiofrequency heating and current drive in the *core* of axisymmetric toroidal magnetic fusion devices. The most serious gap is the lack of a predictive understanding of the amount of power that can be coupled into a fusion plasma with a given launcher design. The negative impacts of this knowledge gap are clear: (i) significant and variable loss of power in the edge regions of confined plasmas and surrounding vessel structures adversely affects the core plasma performance and lifetime of a device; (ii)

the launcher design is partly "trial and error," with the consequence that launchers almost always have to be reconfigured after initial tests in a given device, at an additional cost. Recent JET experience with the new ITER-like antenna has included in excess of 100 shifts of machine operation to learn how to operate it. The second most serious gap is a quantitative lack of understanding of the combined effects of nonlinear wave-plasma processes, energetic particle interactions and non-axisymmetric equilibrium effects on determining the overall efficiency of plasma equilibrium and stability profile control techniques using radiofrequency waves. This is complicated by a corresponding lack of predictive understanding of the time evolution of transport and stability processes in fusion plasmas.

Antennas

Ion cyclotron and LH utilization is often limited in present experiments as a result of the antenna performance. In future devices, several additional issues are likely: coupling at long distance; compatibility of high-performance, steady-state discharges and metallic plasma facing components; reliably maintaining coupled power despite load variations; ability to deliver radiofrequency power on demand without burdensome antenna conditioning; and compatibility with the nuclear environment. The development and validation of a simulation tool that can properly handle complicated antenna structure and the relevant plasma physics are critical. The challenge is to couple electromagnetic simulation of the antenna in the plasma edge with the core absorption modeled correctly. This situation is further complicated by nonlinear physics resulting from the strong radiofrequency fields in the SOL. Current devices can contribute to validation of such codes through comparison with experiment. Detailed measurements of the plasma potential modification and localization of erosion and impurity sources will be important.

Sources

CW high-power sources exist for both ICRF and LH applications. The major issue is maximizing tube lifetime. To address long pulse and lifetime issues, a full test stand would require long pulse supplies and sufficient long pulse load capability.

External Components

Both LH and ICRF require external radiofrequency components to split power, maintain phase control, match the source to the antenna structure (for ICRF) and protect the source from transients caused by plasma events. Current research strategy is to isolate the transmitter from the plasma-induced load variations. This has led to a number of different approaches to provide isolation, both passive and active. The passive approaches have gained more acceptance than active load following. The passive approaches, however, tend to trade efficiency and/or flexibility in favor of load tolerance. Accurate predictions of antenna loads would allow development of network solutions that maximize system efficiency and flexibility. Another approach (suitable for ICRF) is to develop active matching networks, such as ferrite-based tuners, with minimal losses that will maximize the delivered power over a range of plasma conditions. A combination of both test stand and experimental demonstration would be required for qualification.

MAGNETS: RESEARCH REQUIREMENTS

Overall Goal: *Understand the engineering and materials science needed to provide economic, robust, reliable, maintainable magnets for plasma confinement, stability and control*.

Future superconducting magnets for fusion applications require improvements in materials and components to significantly enhance the feasibility and practicality of fusion reactors as an energy source. The fusion program should be developing magnet technologies that are specifically focused on substantially lowering the cost and increasing the availability of the magnets required in fusion power systems. The replacement of a failed toroidal field coil or a major poloidal field coil in a DEMO or fusion reactor is considered to have such an impact on reactor down time (several years) and economics that the system has to be designed to make this a non-credible event. There are primarily three ways in which advances in magnet technology can lower the cost of experiments and fusion power production: 1) by providing conductor and magnet performance, which substantially increases or optimizes the physics performance to allow a smaller or simpler device, e.g., increased magnetic field or some special magnetic field configuration; 2) by lowering the cost of the superconductor and magnet components and/or assembly processes; and 3) by optimizing the configuration of the magnet systems, so that the cost of other fusion subsystems may be reduced.

Opportunity

An integrated program of advanced magnet R&D focuses on developing High Temperature Superconductor (HTS) materials and magnet systems, which offer enormous potential for magnetic fusion energy research experiments, and potentially transformative technological innovation. Accomplishment of a program that fulfills the research gaps and needs described here can potentially revolutionize the design of magnetic fusion devices for very high performance in compact devices with simpler maintenance methods and enhanced reliability.

Research Requirements:

The major question that needs to be answered is: "Can high-temperature superconductors and other magnet innovations be exploited to advance fusion research?" We clearly believe the answer is "yes," but there are substantial development and experimental steps that must be taken to prove it. To achieve this goal, major improvements can be made in the following components:

1) Superconducting wires and cables
2) Mechanical support structure
    (a) External
    (b) Conductor
3) Insulation
4) Joints
5) Quench detection and instrumentation

In item 2, we distinguish between external magnet structure (e.g., a structural case or plate supporting a winding) and structure integral with the conductor, e.g., the conduit material for a cable-in-conduit-conductor (CICC). If important quantitative and achievable goals can be realized for all of these components individually, it should be possible to reduce the cost and perhaps the complexity of fusion devices dramatically. In the following sections we describe issues, opportunities and goals for a few of these important aspects of superconducting magnets and components. In addition, better manufacturing techniques and system integration techniques, for example, using rapid prototyping techniques or demountable superconducting magnets, could help decrease costs through improved manufacturability, reliability and maintainability. We focus on application of HTS to fusion magnets because this is judged to have the largest future impact on machine performance and operation.

HTS Material and Cable Design

Despite their great promise, high-temperature superconductors are still a young technology. The limits on high-temperature superconductors that reflect the early stage of their development are fivefold, including:

1) Cost
2) Performance
3) Piece length
4) Strength
5) Production capacity

Although high-temperature superconductors are not yet a sufficiently established industry to provide conductor for the most demanding fusion applications, the rate of progress in performance is impressive. This is especially true for Yttrium barium copper oxide (YBCO), which is a material of enormous promise for high-temperature and high-field applications. *This is a revolutionary material with the potential for raising field, current density, and temperature simultaneously, while lowering refrigeration requirements.* Achievement of these goals would offer a realistic vision for making an economical future commercial fusion reactor. The vision that high-temperature superconductors are to be used in ultrahigh field magnets should be developed over the moderate-term. Even now, however, the properties and piece lengths are being commercially produced in a range possible for use in low-field fusion devices, e.g., an ST, or with nonplanar coils, e.g., helical or stellarator.

Since most of the commercially produced HTS tapes are made for electric power utility applications, closer involvement of the fusion magnet community with the HTS manufacturers could result in wires that are more amenable to high-current conductor fabrication. The goal of a high-temperature superconductor research program is the production of high effective-current density strands in long lengths and the cabling of ever-larger numbers of strands until the 30-70 kA levels needed by magnetic fusion are attained.

Cabling of strands or wrapping tapes about a core can increase the effective ampere-meters of an unjointed conductor by orders of magnitude, as demonstrated by recent high-voltage transmission line HTS cable demonstration projects, which use multiple tapes wrapped about a cylindrical former/coolant line. This approach has too low an overall current density for a fusion magnet. One central purpose of a fusion conductor and magnet development program would be to develop conductor concepts such as CICC with an adequate combination of current density, field, and cost at reasonably elevated temperature.

An even more important question to answer is whether YBCO conductors could be manufactured as round, multifilamentary wires. For example, what stands between the present YBCO tape configuration and multifilament BSCCO-2212 round wires? The underlying physics of all HTS is supposed to be similar. Although such a breakthrough seems faraway, the resulting benefits would be so valuable that resources should be applied to address the issue.

The superconducting cable design needs to address several key requirements, including (a) high engineering current density, (b) minimal strain degradation, (c) proper stabilization against quenching, (d) reduction of the maximum temperature in case of a quench, (e) low AC losses and (f) efficient cooling.

For HTS tapes, if relevant conductors can only be made as thin, flat tapes, better methods must be developed to produce compact, high-current density cables from this nonideal geometry. Some progress has been made assembling cables using the Roebel pattern from the flat tapes. This serves to increase the overall current capacity, but still has several drawbacks. Some of the expensive superconducting material is lost in the zigzag cutting process and the cable requires development of special machinery to weave the tapes together. Although cables of several kAs can be fabricated this way, they are still about an order of magnitude too low in current for large-scale fusion magnet applications.

Alternatively, better ways to integrate the HTS tapes with the structure, insulation, and cooling of the magnet should be explored. The requirements need to be determined for magnet protection under those circumstances, with conductors operating at higher temperature and well cooled.

Structural Material

Structural materials are required to contain the Lorentz loads of the magnetic pressure vessel, and to contain pressurized helium in cooling tubes or in a CICC superconductor jacket. This strength is required to contain quench pressures, to support gravitational loads, and to maintain coil position and field accuracy. Structural materials must also avoid excessive rigidity in the wrong locations to avoid excessive strains and stresses during assembly, cool down, powering or quench heating. They must be compatible with coil fabrication, and, where applicable, with winding separation for adding insulation after heat treatment, and compatible with conductor conduit heat treatment for CICC wind and react fabrication. With the use of HTS conductors as flat tapes, it is now reasonable to consider integration of the conductor directly into a plate-type structure. These could be done as either layered shells or radial plates. Fabrication methods for these plates must be developed along with methods to integrate the HTS tapes and the turn and layer insulation.

Insulation

Electrical insulation is needed to prevent leakage current and arcs due to magnet voltages during charging, discharging, and quenching. This is required for any type of magnet, including low-temperature superconductors and conventional conductors such as copper. The insulation must be able to withstand repeated voltages that for ITER will be as high as 29 kV. The insulators must also act as a key structural element in maintaining winding pack stiffness and be compatible with local expansion, strain sharing, and load bearing in a conductor-in-plate design. Where insulators can develop tensile loads, they must have adequate shear strength to prevent tearing. In the front layers of the field magnets closest to the plasma, insulation must also be able to withstand neutron and gamma irradiation. The ability to withstand this radiation is frequently the magnet limit that determines the thickness of the neutron shield, and typically dictates the operating life of the magnet systems.

A good insulation system should exhibit four main characteristics:

1) Higher specific dielectric performance in the insulation.

2) Compatibility with heat treatment and other magnet fabrication processes.

3) Ease of application, e.g., impregnation temperature, pot life, etc.

4) Radiation resistance over the design life.

Required research includes the development of inorganic insulating systems and ceramic insulators, and would likely need the coordinated efforts of universities, national labs, and industry. In addition, better means of applying the insulation are required. The insulating step in conventional superconducting winding is tedious. It needs to be carried out with a high level of inspection because of the potential for damaging the superconductor or the insulation itself being damaged during coil assembly and handling. Alternative approaches to the insulation materials, such as new nanodielectric materials and means of integrating the insulation process with the coil manufacturing, should be explored.

Joints

Joints between very large, multi-strand cables of the type required for fusion applications are difficult to make. They must simultaneously achieve the conflicting goals of low resistance, low-AC loss, and high stability. A joint is "superior" if it concurrently improves the goodness factors for volume, DC loss, and AC loss.

Although the fusion magnet community has significant experience producing high-current joints with large cables made from round wires (LTS), there is no equivalent experience in joining large cables or conductors made from many thin, flat HTS tapes.

The fabrication of high-current HTS samples should be developed in the laboratory, with a structured program for understanding joining methods, DC resistance and interface resistances, current transfer, AC losses, and stability. The joint samples should be tested as hairpins and insert coils to establish overall properties. Simple resistance tests can be performed relatively easily with existing equipment. Full-scale prototype joint samples can be tested in the Pulse Test Facility after undergoing some modification to change the test environment from forced-flow supercritical helium to either liquid nitrogen coolant or intermediate temperatures by cooled helium gas.

The greatest programmatic impact will derive from developing a method of joining entire coil cross-sections as a unit while having the ability to be connected, disconnected, and reconnected multiple times with no degradation. This would enable superconducting research facilities in which major components could be readily tested and replaced, and enhance maintainability, availability and inspectability of a DEMO. This is a nontrivial task, since not only should there be excellent electrical connection, but also structural, cooling, and insulating connection. Significant resources are thus warranted to achieve this challenging goal.

Quench Detection and Instrumentation

Quench detection is the Achilles' heel of a superconducting magnet in an erratic pulsed field environment. The specific weakness of tokamak magnets is the plasma disruption, which is unscheduled and varying, making it impossible for its signal to be completely zeroed out predictively. Arbitrary reliability can be built into the power supply interrupters through series connections and redundancy. However, this is much harder to do for quench detectors, which are built into the coils with signal and noise ratios that are intrinsic properties of the sensors. A simplified way of stating the problem is that the magnet voltages are on the order of 10 kV, while the quench signals desirable to detect are on the order of 100 mV, implying a need for five orders of magnitude in noise rejection. Various methods of quench detection needing further development include balanced voltage taps on coil segments, co-wound voltage taps for intrinsic inductive signal cancellation, and co-wound optical fibers that can sensitively measure temperature and strain over a wide range of operating conditions.

The following goals should be achieved:

- Demonstration of standoff voltages in voltage sensors of > 500 V, including helium infiltration over a range of partial pressures and over a range of quench/dump temperatures.
- Demonstration of leak-tightness of sensor extraction ports over experiment-relevant number of cool-down cycles. Goal should be a leak-tightness of < 10^{-6} Torr after 50,000 cycles.
- The intrinsic strain rejection of a fiber-optic thermometer, before filtering and advanced signal processing, should be reduced by a factor of 10^3, over the range of 4 K-150 K. In other words, a change in the conductor strain of 0.1 % should change the effective differential optical path by no more than 1 ppm. The higher the critical temperature of a superconductor, the more attractive fiber optic thermometry becomes for quench detection, because of the rapidly improving temperature sensitivity of the index of refraction.
- Demonstrate feasibility of commercial fiber use with an HTS magnet.

Prototype Magnet Development

Although lab scale tests and component development can lead to viable solutions for the issues discussed above, integration of these components into a magnet is nontrivial, and may lead to complications and synergistic effects, resulting in a magnet that is unable to achieve all its design goals. Therefore, once the OFES strategic planning process identifies a next-step US device, the design goals should be then focused on those required by the device concept. Then all development steps can be proven on a relevant-scale prototype coil or coils, and tested under full-scale operating requirements. Depending on the scale of the magnet, existing facilities for testing should be used or modified to carry out the test program.

R&D Strategy

A number of critical technology areas have been identified to reduce the cost, increase the performance, and improve the reliability of superconducting magnets for fusion applications. The specific goals and criteria outlined here form the basis of an R&D program that should be supported through a significant expansion of the present, very modest, enabling technologies magnet program. This will require coordinated efforts by universities, national laboratories, and industry. A reasonable program structure would include a distribution of efforts ranging from lab-scale R&D, prototype component development, full-size magnet tests, and eventually incorporation into a next-step device. By this we mean that any next-step fusion experiment constructed in the US should strongly consider using the best available superconducting magnet technology as a viable option for enhancing the mission of the device.

Connections between Research Requirements and ReNeW Research Thrusts

The many research requirements identified for this Theme will be addressed by the set of Research Thrusts. In many cases, the various research activities from a given panel are included in multiple thrusts. These are summarized in Table 3, with the thrusts for each panel listed in approximate order of their connection to the panel. Detailed descriptions of each thrust are given in Part II of this Report.

Panel	**Research Thrust**	**Comments**
Measurement	**Thrust 1**: Develop measurement techniques to understand and control burning plasmas.	Primary thrust for new and robust BP diagnostics.
	Thrust 6: Develop predictive models for fusion plasmas, supported by theory and challenged with experimental measurement.	New measurements to validate models.
	Thrust 5: Expand the limits for controlling and sustaining fusion plasmas.	Measurements and analysis to enable real-time control.
Integration of steady-state, high-performance burning plasmas	**Thrust 8:** Understand the highly integrated dynamics of dominantly self-heated and self-sustained burning plasmas.	Integration of high-gain, steady-state burning plasma core.
	Thrust 12: Demonstrate an integrated solution for plasma-material interfaces compatible with an optimized core plasma.	Integration of sustained core plasma with high heat flux boundary.
	Thrust 5: Expand the limits for controlling and sustaining fusion plasmas.	Develops needed control tools and establishes performance limits.
Validated theory and predictive modeling	**Thrust 6**: Develop predictive models for fusion plasmas, supported by theory and challenged with experimental measurement.	Primary thrust, with application to all ReNeW Themes and many thrusts.
	Thrust 9: Unfold the physics of boundary layer plasmas.	Focused on scrape-off layer models and solutions.
	Thrust 3: Understand the role of alpha particles in burning plasmas.	Focused on alpha physics, primarily in ITER.

Panel	Research Thrust	Comments
Control	**Thrust 5**: Expand the limits for controlling and sustaining fusion plasmas.	Primary thrust.
	Thrust 8: Understand the highly integrated dynamics of dominantly self-heated and self-sustained burning plasmas.	Provides tests and integration in burning, self-sustained regimes.
	Thrust 2: Control transient events in burning plasmas.	Disruption and ELM control.
Transient Plasma Events	**Thrust 2:** Control transient events in burning plasmas.	Primary thrust, with strong ITER focus.
	Thrust 5: Expand the limits for controlling and sustaining fusion plasmas.	Issues and solutions for steady-state, high-performance tokamaks.
	Thrust 17: Optimize steady-state, disruption-free toroidal confinement using 3-D magnetic shaping, and emphasizing quasi-symmetry principles.	Issues and solutions for steady-state stellarators.
	Thrust 3: Understand the role of alpha particles in burning plasmas.	Fast alpha loss events.
	Thrust 11: Improve power handling through engineering innovation.	Transient event tolerant walls.
Plasma Modification by Auxiliary Systems	**Thrust 5**: Expand the limits for controlling and sustaining fusion plasmas.	Primary thrust, develops heating, current drive, fueling needed for active control and sustainment.
	Thrust 4: Qualify operational scenarios and the supporting physics basis for ITER.	Addresses auxiliary system issues for ITER scenarios.
	Thrust 6: Develop predictive models for fusion plasmas, supported by theory and challenged with experimental measurement.	Physics understanding of waves, transport.
	Thrust 9: Unfold the physics of boundary layer plasmas.	Addresses issues of RF waves in boundary.
Magnets	**Thrust 7:** Exploit high-temperature superconductors and other magnet innovations to advance fusion research.	Primary thrust, with application to multiple Themes and panels.

Table 3. Connections Between Theme 2 Requirements and Research Thrusts.

It can be seen from the above table that the mapping between panels and Thrusts is complex. This reflects the strong interrelation between issues arising in different panels and Themes. In some cases, it was judged preferable to attack groups of related issues together. The most notable example of this is Thrust 5. Controlling any aspect of plasma behavior, including transient events, requires accurate sensors and effective auxiliary systems, as well as state-of-the-art algorithms based on physical models. This broad Thrust takes an integrated approach to the control and sustainment challenge, and combines key elements of the Measurements, Control, Auxiliary Systems, Transient Plasma Events and Validated Theory and Predictive Modeling panel research requirements.

In several other cases, subsets of issues from a given panel are addressed in different thrusts, often combined with other themes, to form focused and coherent research activities with reduced overlap. For example, the extremely broad issues arising in integration of steady-state, high-performance burning plasmas are covered by two thrusts, with Thrust 8 focusing on the new challenges of dominantly self-heated core burning plasmas, and Thrust 12, joint with Theme 3, assessing integrated solutions of sustained core plasmas with DEMO-level heat fluxes. Due to its critical importance, multiple Thrusts also address the issue of avoiding transient events. Thrust 2 focuses on issues and solutions for disruptions and ELMs relevant to ITER, while Thrust 5 addresses the even greater challenges, and broader range of potential solutions, for steady-state, higher performance plasmas. Fast particle loss events, given their strong connection to alpha physics, are treated in Thrust 3. Activities to assess issues and solutions involving stellarator geometries are in Thrust 17, led by Theme 5, while novel PFCs that may better tolerate transient events will be developed and assessed in Theme 3, Thrust 11, and integrated in Thrust 12, which is joint with Theme 3. The ways in which the issues covered in this Chapter will be addressed by proposed activities of the Research Needs Workshop are discussed in the descriptions of *all* of the Thrusts in the table, found in Part II of this Report (Thrusts 6-10).

SUGGESTIONS FOR FURTHER READING

1. *"Priorities, Gaps and Opportunities: Towards A Long-Range Strategic Plan for Magnetic Fusion Energy" Report to FESAC Oct 2007, chaired by M. Greenwald.*

2. *Final Report–Workshop on Burning Plasma Science: Exploring the Fusion Science Frontier* (Fusion Energy Sciences Advisory Committee, 2000). http://fire.pppl.gov/ufa_bp_wkshp.html
3. *Burning Plasma: Bringing a Star to Earth* (National Academy of Science, 2004).

4. *Plasma Science: Advancing knowledge in the national interest*, Plasma 2010 Committee, Plasma Science Committee, National Research Council (National Academies Press, 2007).

5. C. Gormezono *et al*, *Progress in the ITER Physics Basis, Chapter 6: Steady state operation,* Nuc Fus. **47** (2007) S285.

6. F. Najmabadi and the Aries Team, "*Overview of ARIES-RS tokamak fusion power plant," Fusion Engineering and Design* **41** *(1998) 365-370, and other articles in this issue.*

7. F. Najmabadi and the Aries Team, *"The ARIES-AT advanced tokamak, Advanced technology fusion power plant*," Fusion Engineering and Design **80** (2006) 3-23, and other articles in this issue.

CREATING PREDICTABLE, HIGH-PERFORMANCE, STEADY-STATE PLASMAS THEME MEMBERS

MEASUREMENT

(Joint with Theme 1 Panel "DIAGNOSING A SELF-HEATED PLASMA")
JIM TERRY, Massachusetts Institute of Technology (Panel Leader)
REJEAN BOIVIN, General Atomics (Panel Deputy Leader)
MAX AUSTIN, The University of Texas at Austin
TED BIEWER, Oak Ridge National Laboratory
DAVID BROWER, University of California, Los Angeles
DANIEL DEN HARTOG, University of Wisconsin–Madison
WILLIAM D. DORLAND, University of Maryland, College Park
DAVID JOHNSON, Princeton Plasma Physics Laboratory
GEORGE McKEE, University of Wisconsin–Madison
TONY PEEBLES, University of California, Los Angeles
DAN STUTMAN, Johns Hopkins University
KEN YOUNG, Princeton Plasma Physics Laboratory (Retired)

INTEGRATION OF STEADY-STATE, HIGH-PERFORMANCE BURNING PLASMAS

CHARLES KESSEL, Princeton Plasma Physics Laboratory (Panel Leader)
DAVID ANDERSON, University of Wisconsin–Madison
PAUL BONOLI, Massachusetts Institute of Technology
ANDREA GARAFALO, General Atomics
BRUCE LIPSCHULTZ, Massachusetts Institute of Technology
DALE MEADE, Fusion Innovation Research and Energy
MASANORI MURAKAMI, Oak Ridge National Laboratory
FARROKH NAJMABADI, University of California, San Diego
PETER POLITZER, General Atomics

VALIDATED THEORY AND PREDICTIVE MODELING

GEORGE TYNAN, University of California, San Diego (Panel Leader)
ROBERT BUDNY, Princeton Plasma Physics Laboratory
TROY CARTER, University of California, Los Angeles
C.S. CHANG, New York University
MARTIN GREENWALD, Massachusetts Institute of Technology
ARNOLD KRITZ, Lehigh University
WILLIAM TANG, Princeton Plasma Physics Laboratory
PHILIP SNYDER, General Atomics

CONTROL

(Joint with Theme 1 Panel "CONTROLLING AND SUSTAINING A SELF-HEATED PLASMA")

DAVID HUMPHREYS, General Atomics (Panel Leader)
JOHN FERRON, General Atomics (Panel Deputy Leader)
TOM CASPER, Lawrence Livermore National Laboratory
DAVID GATES, Princeton Plasma Physics Laboratory
ROB LAHAYE, General Atomics
TOM PETRIE, General Atomics
HOLGER REIMERDES, Columbia University
ALAN TURNBULL, General Atomics
MIKE WALKER, General Atomics
THOMAS WEAVER, Boeing
STEVE WOLFE, Massachusetts Institute of Technology

TRANSIENT EVENTS

(Joint with Theme 1 Panel "MITIGATING TRANSIENT EVENTS IN A SELF-HEATED PLASMA")

RICHARD HAWRYLUK, Princeton Plasma Physics Laboratory (Panel Leader)
STEVE KNOWLTON, Auburn University (Panel Deputy Leader)
JON MENARD, Princeton Plasma Physics Laboratory (Panel Deputy Leader)
ALLEN BOOZER, Columbia University
ERIC FREDRICKSON, Princeton Plasma Physics Laboratory
BOB GRANETZ, Massachusetts Institute of Technology
VALERIE IZZO, University of California, San Diego
EDWARD J. STRAIT, General Atomics
JOHN WESLEY, General Atomics
DENNIS WHYTE, Massachusetts Institute of Technology

PLASMA MODIFICATION BY AUXILIARY SYSTEMS

J. RANDALL WILSON, Princeton Plasma Physics Laboratory (Panel Leader)
LARRY BAYLOR, Oak Ridge National Laboratory
LEE BERRY, Oak Ridge National Laboratory
JOHN GOETZ, University of Wisconsin–Madison
LARRY GRISHAM, Princeton Plasma Physics Laboratory
MIKLOS PORKOLAB, Massachusetts Institute of Technology
RON PRATER, General Atomics
ROGER RAMAN, University of Washington
RICHARD TEMKIN, Massachusetts Institute of Technology

MAGNETS

JOSEPH MINERVINI, Massachusetts Institute of Technology (Panel Leader)
LESLIE BROMBERG, Massachusetts Institute of Technology
MICHAEL GOUGE, Oak Ridge National Laboratory
DAVID LARBALESTIER, Florida State University
BRAD NELSON, Oak Ridge National Laboratory
GIAN LUCA SABBI, Lawrence Berkeley National Laboratory
JOEL SCHULTZ, Massachusetts Institute of Technology
RICHARD THOME, General Atomics

THRUST LEADERS

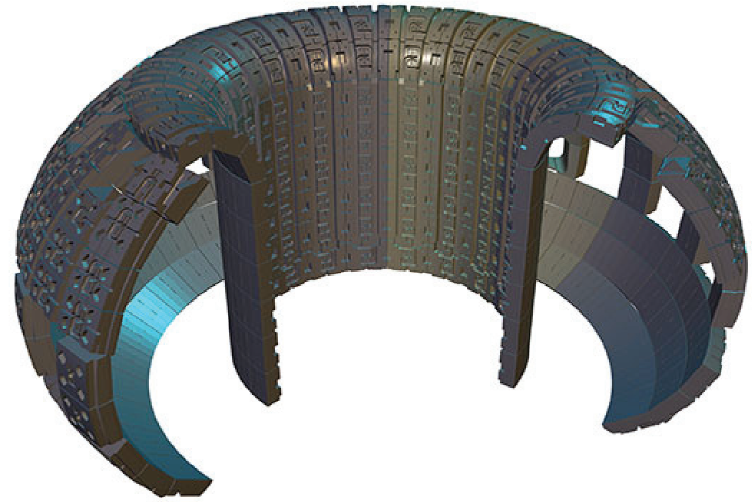
Thrust 5: Expand the limits for controlling and sustaining fusion plasmas.
ALAN TURNBULL, General Atomics
Thrust 6: Develop predictive models for fusion plasmas, supported by theory and challenged with experimental measurement.
MARTIN GREENWALD, Massachusetts Institute of Technology
Thrust 7: Exploit high-temperature superconductors and other magnet innovations to advance fusion research.
JOSEPH MINERVINI, Massachusetts Institute of Technology
Thrust 8: Understand the highly integrated dynamics of dominantly self-heated and self-sustained burning plasmas.
CHARLES KESSEL, Princeton Plasma Physics Laboratory

THEME LEADERS

AMANDA HUBBARD, Massachusetts Institute of Technology, Chair
CHARLES GREENFIELD, General Atomics, Vice-Chair
MARK FOSTER, Office of Fusion Energy Sciences, U.S. Department of Energy

THEME 3:

TAMING THE PLASMA-MATERIAL INTERFACE



ON PREVIOUS PAGE
Design of the ITER firstwall system: shielding blanket (top) and divertor (bottom).

THEME 3: TAMING THE PLASMA-MATERIAL INTERFACE

Introduction

SCOPE AND FOCUS

The hot plasma in the core of a fusion energy device must interact with low-temperature material walls. That interaction is mediated by a thin plasma layer called the scrape-off layer (SOL). Many of the recent improvements in plasma core performance have been enabled by either improved understanding of the SOL or improved plasma facing components (PFCs). The report of the Priorities, Gaps and Opportunities Panel, or "Greenwald Report"[1], identified three major issues at the plasma-material interface. We must "understand and control all of the processes which couple plasma and nearby materials" (issue 8). We must "understand the materials and processes that can be used to design replaceable components which can survive the enormous heat, plasma and neutron fluxes without degrading the performance of the plasma or compromising the fuel cycle" (issue 9). We must "establish the necessary understanding of plasma interactions, neutron loading and materials to allow design of RF [radiofrequency] antennas and launchers, control coils, final optics and any other diagnostic equipment which can survive and function within the plasma vessel" (issue 10). A total of nineteen gaps in knowledge was identified for the three issues.

At least a dozen new facets of scrape-off layer behavior have been uncovered over the past decade. This improvement in understanding has been accomplished through a combination of new diagnostics (edge Thomson scattering, fast camera imaging), improved edge models (coupled fluid and neutral codes), and more dedicated plasma operation for edge diagnosis. Several of these new facets have caused a major alteration to the design of ITER. The remaining uncertainty in the scaling of the power scale-length in the scrape-off layer has placed very tight constraints on the design of ITER components.

Plasma facing components have increased significantly in heat flux capability in the last decade due to new materials (carbon fiber composite), improved design (protection of leading edges and better shaping relative to plasma flux surfaces), surface conditioning (boronization, Helium glow cleaning, high-temperature baking), and in some cases active cooling. Laboratory studies and limited application to fusion devices have proven the capability of water-cooled plasma facing components to achieve the heat removal required for ITER. There is an ongoing debate about the choice of plasma facing materials for ITER with some advocating an all tungsten option while others argue for all graphite, but the baseline PFCs for deuterium-tritium (D-T) operation have only beryllium and tungsten.

Long pulse radiofrequency antennas and launchers have been operated at high power on the Tore Supra device where pulse lengths are measured in minutes. Improvements in feedback control have yielded more stable radiofrequency heating in existing devices. Such operation has revealed interesting new physics in the plasma edge. Electron cyclotron resonance heating has been used

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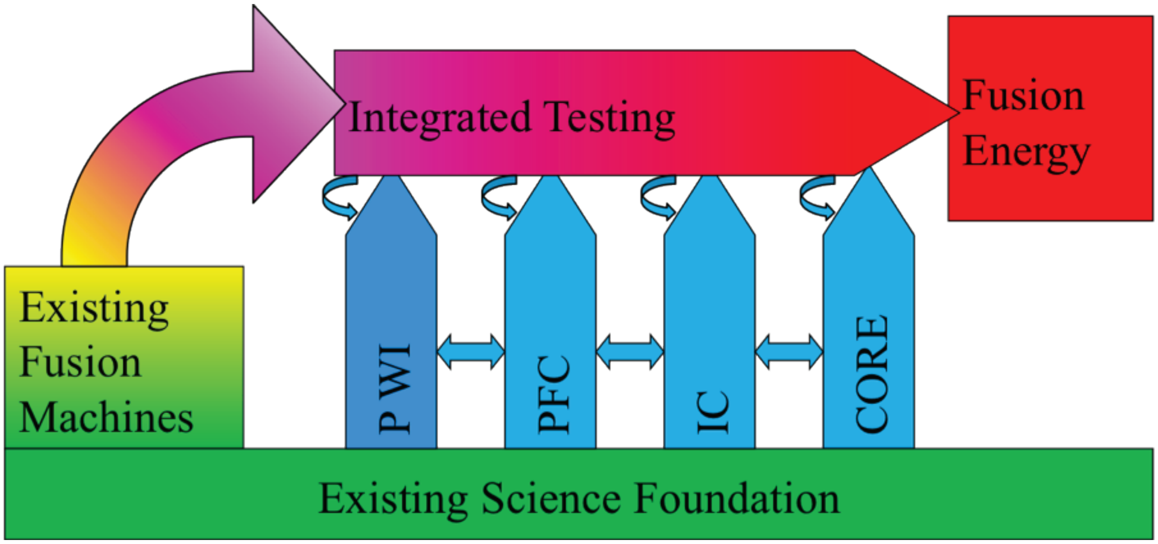
[1] *"Priorities, Gaps and Opportunities: Towards A Long-Range Strategic Plan For Magnetic Fusion Energy" Report to FESAC Oct 2007, M. Greenwald, Chairman.*

to stabilize deleterious plasma modes and improve plasma performance. The key to this ability is an improved final steering mirror exposed to the plasma. ITER is supporting the development of several new diagnostics that are able to operate in the neutron environment in ITER. While the developments needed for ITER have driven necessary research, there is still a big gap to a DEMO device.

While the progress in taming plasma-material interactions is notable, there is still a large gap to close before we are ready to undertake a DEMO. Greater emphasis needs to be placed on edge plasma diagnostics and development of physics models of the edge on existing machines. To meet the expected conditions in future fusion machines, existing laboratory devices for basic plasma-material interaction studies need to be upgraded and new facilities built. Innovative solutions that can increase the capability and robustness of plasma facing components require a new investment of funds and people. These changes should be coordinated on a national level. Without these changes, the US Fusion Energy Sciences Program will not have the knowledge or skills to advance to the next generation of fusion machines, let alone be ready to participate in DEMO. As illustrated below, several research thrusts must cooperate in parallel to build the foundation for an integrated experiment leading to a DEMO-type device.



Research Requirements

The Taming the Plasma-Material Interface participants have identified the following key requirements and high-leverage candidate investments.

PLASMA-WALL INTERACTIONS

Plasma-wall interactions (PWI) encompass scientific issues that are among the most critical for fusion power, affecting: 1) Lifetime of PFCs, owing to sputtering and transient erosion, 2) Plasma contamination by eroded material, 3) Tritium co-deposition in eroded and redeposited material, and 4) Operating limits on core plasma (beta, confinement, edge temperature and density, duty factor, etc.). Central to these issues is the science of plasma-material interactions (PMI). As summarized in the white papers, there are major gaps in modeling, analysis, and understanding PMI phenomena in existing machines, ITER, and certainly post-ITER devices.

DIAGNOSTIC INVESTMENTS

Can diagnostics be improved to identify the missing physics that control plasma-wall interactions?

Present edge/SOL diagnostic capability is seriously inadequate and is the main impediment to the identification of missing edge physics. On most tokamaks today, only n_e and T_e (without energy distribution information and usually only time-averaged) are measured regularly — at a few locations. A spatially extensive set of edge measurements is required. Understanding aerodynamic lift would have been impossible if the air velocity was measured at just one or two locations around airfoils; the SOL is much more complicated and present spatially sparse diagnostic sets cannot identify edge controlling physics. Extensive sets of pop-up probes could be used to map out the 4-D distributions of n_e, T_e, ϕ ,$v_{||}$, $v_{\perp}$, T_i and electron and ion energy distributions. Thomson scattering with a high dispersion instrument such as a Fabry-Perot could measure electron and ion energy distributions and Z_{eff}. A multi-laser Thomson system (firing laser 1, 2, 3 in rapid succession) could make images that follow the time evolution of "blobs" and edge localized modes (ELMs). Charge exchange recombination (CER) spectroscopy using new powerful pulsed ion diode neutral beams (10^6 A/m^2, 1 μs, 125 keV/amu) may permit measurement of T_i and $v_{||}$ into the separatrix. Real time, *in situ* surface diagnosis of hydrogen-uptake, erosion, deposition, co-deposition, etc., is likewise required, using various proposed concepts to perform surface analysis inside the vessel. Better characterization of SOL cross-field transport is needed. Since most of these techniques are labor-intensive, a significant increase in the number of edge diagnosticians will be required.

Off-line facilities are needed to provide **in-situ surface dynamic diagnosis** of innovative materials to quickly test fundamental properties that can be accelerated for further testing in more complex environments (e.g., linear plasma device and fusion device). We must also close a gap between materials modeling and experimental techniques to elucidate the fundamental damage mechanisms in the complex plasma edge environment and the particle-induced material evolution during steady-state and transient plasma events. In particular, we must study how the PWI at length scales less than a micron couples to structural effects at depths of a few microns or more. Off-line facilities can also introduce appropriate energetic particle beams to study the bulk and surface interface to understand effects on retention, diffusion, segregation, permeation and phase transformation, among other mechanisms.

DEDICATED EXPERIMENTAL TIME

Can more dedicated experimental time on existing facilities or a dedicated new facility provide the edge measurements required to sort out key plasma-wall interaction issues?

There is a strong need for greater predictability of the plasma scrape-off layer's basic parameters and PWI issues. We critically need data to make design choices for both a D-T Component Test Facility (CTF) and DEMO. The most fundamental requirement is for a predictive-quality scaling of the power scrape-off width. This has a strong impact on the heat-flux handling capability required of the divertor in future devices. Carefully planned experiments on existing facilities are needed, with controls over radiative power fraction in the plasma and in the SOL, as well as monitoring the degree of divertor recycling and/or detachment. Other high-level issues that require more thorough study include the dependencies on key parameters (major/minor radii, magnetic field, and discharge current and power) of the density and/or fueling rate at which the transitions occur from sheath-limited to high recycling conduction-limited, and then to detached operation. It is well known that the particle equilibration time in the SOL and material surfaces can be much longer than the (solid) thermal equilibration time in these systems. Thus, much longer pulses will be required in a future device designed to understand and control the dynamics of high recycling and detachment. It is also anticipated that the heat flux in a CTF or DEMO will far exceed what can currently be studied for long pulses in tokamak experiments. The temperature of the plasma facing components will also be much higher than in current experiments. Together these will strongly change the dynamics of plasma recycling. Thus, the physics, particularly of high recycling and detachment, may be quite different from what is encountered in current experiments.

Therefore, an intermediate-scale experiment capable of providing the hot walls, long pulse, and high-power plasma-material interface of CTF and DEMO, with high flexibility to test innovations and with excellent diagnostic access, is needed to validate the understanding of plasma-wall interactions at a level that supports a major long-pulse, high-power D-T initiative.

IMPROVED MODELS AND CODE COMPONENTS

Can basic scrape-off layer plasma models and code components be improved to validate predictions for future devices?

Plasma transport in the edge/SOL has a direct impact on plasma energy and particle fluxes to PFCs and on impurity transport and redeposition. The observed nature of this transport has long been differentiated from that in the core plasma, with historical measurements of large density fluctuations relative to the time-average values, which can approach unity in the SOL. More recent measurements have shown additional effects such as strong intermittency, filamentation, toroidal asymmetry, and large flows.

Two-dimensional plasma-fluid transport codes (e.g., UEDGE, B2-Eirene, EDGE2D, OSM-Eirene) are the primary tools used to model particle and heat fluxes in a tokamak boundary layer. Up to now such codes have not provided full predictive capability for the scrape-off layer profiles, but typically fit data at one location and compare at remote locations. Yet even with this modest goal, it has been difficult to reproduce the observed plasma phenomenology, such as very strong (near-sonic) plasma flow around the plasma periphery, without postulating beforehand strong spatial dependencies of the transport coefficients. Moreover, experiments clearly show that simple diffusive and/or convective transport descriptions are not appropriate. Large amplitude fluctuations, critical-gradient transport dynamics and intermittent transport events involving quasi-coherent structures (blobs, ELMs) dominate the edge plasma. Accurate descriptions of divertor detachment are also lacking. Likewise, the present generation of edge plasma turbulence simulation codes (e.g., BOUT, DALFTI, ESEL) appears to reproduce much of the edge plasma phenomenology (e.g., blob-like propagation, intermittency), but there are only limited and approximate comparisons of wave number spectra, radial fluxes, and blob speeds.

Our present deficiencies in boundary layer physics understanding are a major obstacle for accurately projecting to edge plasma conditions in ITER and DEMO – devices which will have combinations of power densities, plasma collisionalities and neutral opacities that are inaccessible in today's devices. Current tokamaks will not be able to access the high-power output per unit surface area of ~1 MW/m^2 that is required for a DEMO-class reactor, nor will they be able to simultaneously attain the divertor collisionality and neutral opacity of a DEMO. Thus, to understand edge plasma behavior in regimes that more closely match those of a DEMO and to validate corresponding edge plasma models, a new high-plasma heat-flux tokamak facility will be needed. Once the underlying edge/SOL instabilities are identified, a key need is to develop an edge/SOL transport model for the intermittent blob/ELM plasma filaments. Furthermore, this intermittent transport likely causes inward convection of impurities that must be understood.

CODE DEVELOPMENT

Can simulation codes be developed with improved coupling among plasma components (neutral, plasma, and impurity transport) and overall material response to yield predictive capability?

As discussed, these are extensive gaps in existing plasma-material interaction theory, modeling/ code efforts and experimental validation. Gaining understanding and predictive capabilities in this critical area will require addressing simultaneously complex and diverse physics occurring over a wide range of lengths (angstroms to meters) and times (femtoseconds to days). This will require further development of detailed physics models and computational strategies at each of these scales, as well as algorithms and methods to strongly couple them in a way that can be robustly validated. While present research, combining at most a few of these scales, already pushes the state-of-the-art in technique and available computational power, simulations spanning the multiple scales needed for ITER and DEMO will require extreme-scale computing platforms and integrated physics and computer science advances.

The goal is to develop comprehensive computational models for predictive, self-consistent, validated, and time-dependent plasma-material interactions. This would first encompass modeling of the edge and scrape-off layer plasma, including treatment of turbulent transport, and full coupling of plasma ions and electrons, neutrals, photons, and electromagnetic fields. Next, plasma contamination from near-surface transport of sputtered or vaporized material, and quantification of PFC particle and photon fluxes (and prediction of instability regimes) would be included. A critical related issue is to predict the near-surface material response to the extreme plasma fluxes of photons and particles, both under normal and transient operation. This involves modeling of sputtering erosion and redeposition and other time-integrated PFC processes (e.g., dust formation and transport, helium, and D-T induced microstructure formation and flaking) and the resultant impurity transport, core plasma contamination, mixed-material formation, and tritium co-deposition in redeposited materials. The material and edge plasma response to transient processes such as high-powered ELMs, Vertical Displacement Events (VDEs), plasma disruptions, and runaway electrons would be an important component of this effort.

INNOVATIVE DIVERTORS

Can innovative divertor configurations be developed to control the plasma heat flux problem?

Large, localized plasma heat exhaust is a critical problem for the development of tokamak fusion reactors. Excessive heat flux erodes and melts plasma facing materials, thereby dramatically shortening their lifetime and increasing the impurity contamination of the core plasma. A detailed assessment of the ITER divertor has revealed substantial limitations on the plasma operational space imposed by the divertor performance. For a commercial fusion reactor the problem becomes worse in that the divertor must accommodate 20% of the total fusion power (less any broadly radiated loss), while not allowing excessive impurity production or tritium co-deposition and retention. This is a challenging set of problems that must be solved for fusion to succeed as a power source; it calls for a substantial research investment.

Promising new ideas have been developed for modifications to present-day divertors that may substantially improve their capability to disperse high-heat flux and perhaps provide control of ELMs. These new approaches include the Super X divertor that guides the near-separatrix SOL flux tubes to a larger major radius to increase the surface area available for power deposition, and the Snowflake divertor concept that produces a second-order null in the poloidal magnetic field at the X-point. Engineering innovations to achieve superior target (tile) alignment can make possible the effective use of small incidence angles, lowering peak power deposition. Another promising approach is the use of liquid metal (Li, Sn, Ga) divertor surfaces that can increase the heat-flux capability by flowing the heated material to a non-divertor cooling, and by eliminating most erosion concerns. There are also ideas about flow through solids such as gas-injected carbon or boron. All of these areas are only emerging concepts that require substantially more analysis and definitive experimental tests. Given the need for a large improvement in this area, we advocate a substantial program to systematically assess them.

TRANSIENT IMPACT

Can the impact of transients, long pulse, and elevated temperature operation be controlled?

The impact of transient events on plasma-material interactions with DEMO-relevant plasma facing components is critically important. An unmitigated disruption can significantly erode or damage plasma facing surfaces. In high confinement mode (H-mode) operation, repetitive ELM instabilities expel plasma thermal energy that heats divertor plasma facing component surfaces and limits their service life. Analysis shows that the maximum ELM size that can be tolerated without reaching the threshold energy density of ~1.0 (MJ/m^2) for significant erosion is small in ITER (~10% core plasma energy content). Likewise,VDEs and runaway electrons pose major challenges.

The tritium fuel cycle and vessel inventory is a significant concern for ITER due to potential for co-deposition of tritium with eroded PFC material. In DEMO the issues of the tritium fuel cycle are completely different due to the high-operating temperature. These effects (impact of transients, long pulse, and elevated temperature) can be cost-effectively studied in dedicated, non-toroidal experiments designed to understand the basic PWI processes in a controlled environment. These can, for example, be operated in pulsed mode to simulate disruptions and ELMs to better understand the effects of these phenomena on PWI over longer exposure times than can be currently achieved in today's fusion devices. Similarly, the effect of extended pulse lengths and elevated wall temperature on the PWI process can be addressed in simplified devices, at lower cost, and much sooner, without requiring the complexity of the full tokamak system. Of course, the toroidal effects will also need to be investigated, and new advanced toroidal devices will be required for those purposes. In particular, neither ITER nor the upcoming Asian long pulse tokamaks will have the optimum combination of hot wall, long pulses, high-power density, and high-duty factor capability required to test solutions to these issues in a toroidal system. While the specifications in terms of pulse length, duty factor, fractional operation in deuterium (vs. hydrogen) and both flexibility and accessibility for PMI and core-edge integration studies need to be completed, it is clear there exists an important role in the world program for such devices.

PLASMA FACING COMPONENTS

The ReNeW PFC panel has reviewed the status of high heat-flux removal development, which indicates the difficulties of designing for a maximum heat flux of 10 MW/m^2 (see Doerner[2]) while maintaining the ability to withstand a high-power ELM energy flux of 0.5 MJ/m^2 (see Ritz[3]) when helium is used as the coolant. For the latter case, due to the potential damage from cyclic stresses, the best approach is to avoid pulse loads from high-energy density ELMs. The avoidance of disruptions is mandatory. An important transition is the evolution from mostly inertially cooled tokamak experiments to ITER, where the pulse length will be increased from a few seconds to the range of 400-3000 s, the structural material will be stainless steel, and components will have to be actively water-cooled copper and maintained remotely. For a successful development of PFCs, the following research is required to advance to DEMO.

- Identification and qualification of materials that can take the heat loads, survive damage from edge alphas, neutron flux and fluence at operating surface temperature while maintaining acceptable erosion rates and core plasma contamination levels.
- Properties of low activation solid materials, joining technologies and cooling strategies sufficient to design robust first-wall and divertor components in a high heat flux, steady-state nuclear environment and with adequate design margin.
- Characterization of welds, brazes or other joining techniques that can carry high heat fluxes in the presence of high-edge alphas and neutron flux and fluence.
- Strategies for heat removal with gas coolant at high temperatures while maintaining structural integrity, especially with respect to temperature excursions or other off-normal events.
- Characterization of tritium effects including permeation, embrittlement, retention and migration.
- Strategies for handling dust migration and inventories.
- Strategies for remote maintenance for high machine availability.

INNOVATIVE DESIGN OF THE SOLID SURFACE PFCS

Can innovative solid surface components be developed for DEMO?

ITER will operate with an integrated neutron fluence of 0.3 MW-yr/m^2, and the plasma facing material will consist of a combination of Be, carbon fiber composite (CFC) and tungsten (W). Fusion machines beyond ITER will require the development of robust helium-cooled solid surface PFCs for the first wall to withstand steady-state maximum surface loading of ~ 1 MW/m^2, the divertor

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[2] *R. Doerner, "PMI issues beyond ITER" Presentation at the International HHFC Workshop on Readiness to proceed from near-term fusion systems to power plants, San Diego, CA, USA Dec 10-12, 2008.*

[3] *G. Ritz, T. Hirai, J. Linke et al., "Post-examination of helium-cooled tungsten components exposed to DEMO specific cyclic thermal loads," To be published in Fusion Engineering and Design, 2009.*

to handle 10 MW/m^2, and ELMs with a pulsed energy density of ≤ 0.5 MJ/m^2. Research should include heat flux removal enhancement techniques, e.g., internal roughening, swirl tape, microjets and porous media. Design margin enhancement should also be included. Innovative magnetic configurations for high heat flux reduction, like the Super X and Snowflake divertor configurations, will require development of PFCs compatible with the unique magnetic geometry and possible differences in plasma transport.

Development of solid surface PFC designs that are tolerant to a few off-normal events should continue. This should include low-Z material loaded tungsten surfaces and low-Z material alloying of tungsten-surface materials. These concepts rely on the vapor shielding effects from the low-Z material to protect the tungsten surfaces and be able to withstand a few disruptions. Real time replenishment of the low-Z surface materials will be needed for this option.

Innovative refractory materials with high thermal conductivity compatible with the design of He coolant or liquid surfaces must be developed for neutron fluence up to 15 MW-yr/m^2. Joining techniques compatible with He or liquid metal coolants are needed for the solid surface option to be viable. Tungsten or tungsten alloys are the leading candidates for such material because of their high thermal conductivity, low erosion rate, and high melting point. However, neutron embrittlement is a serious issue the innovative alloys must address.

TESTING REQUIREMENTS TO VALIDATE DESIGN ISSUES

Can testing scenarios be developed for new plasma facing component designs on new or existing laboratory and fusion facilities?

Helium-cooled heat sinks (including fabrication methods) for high heat flux removal capacity refractory material PFCs need to be developed and validated through rigorous testing. The joint between the plasma facing material and the heat sink is typically a few microns thick and composed of intermetallics. (These intermetallics have uncertain properties for which there are no measurements after neutron irradiation.) Extensive heat flux testing of new designs must be conducted on test stands and the designs applied to long pulse fusion devices before they can be considered for DEMO. This development should include different nondestructive inspection techniques. Neutron irradiation typically hardens materials while decreasing ductility and fracture toughness. Development of a database on nuclear irradiation effects on the above advanced and innovative PFC concepts is required to address issues, including cyclic fatigue, thermal creep, fracture toughness, and fracture mechanics at interfaces.

Results from linear machines have shown that with the impingement of edge alphas, blisters can be created at the first-wall chamber owing to trapping of the impinging helium ions, and generation of W-fuzz on the divertor surfaces. No explanation for the cause of the generation of the W-fuzz is available. In general, the damage effects from edge alphas have not been systematically studied. To address the synergistic effects, new and upgraded facilities that can allow flexibility in the testing of PFC surface materials with the combination of simulated neutron, alphas and plasma environment at high temperature, and operate in pulsed and steady-state modes, will be needed.

There is very little experience with steady-state actively cooled PFCs on fusion devices. The database is insufficient to determine the failure mechanisms and the mean time between failures. There is an inadequate safety database regarding the loss of coolant and related accidents. There is also a lack of diagnostics for PFC components on existing devices. Even with improved diagnostics, integrated components testing of advanced and innovative PFCs in a fusion nuclear environment will be required to design for DEMO. Adequate design margins should be demonstrated in the development of PFCs.

Minimization of tritium permeation through the PFC to the coolant must be included in the design of the PFC systems. Some materials (e.g., titanium, vanadium, tantalum) are known to be susceptible to hydrogen embrittlement. Tritium retention in PFCs must be limited to avoid safety issues related to tritium inventory, routine release and release during accidents. Permeation will only be significant for devices that have large tritium throughput or fluence to the PFCs. It can only be measured on components tested in devices like ITER or CTF. Laboratory measurements, including irradiated samples, could provide the fundamental database.

- Helium accumulation at the divertor should be minimized.
- Development of remote handling and maintenance equipment will be needed. At the same time, tailoring the PFCs with precise component alignment should be investigated.
- Development of a safety related database for the CTF and DEMO designs will be required to satisfy licensing authorities.
- All of the above will need to be supported by integrated modeling codes to generate predictive capability on future PFC designs.

LIQUID SURFACE DEVELOPMENT

Can practical liquid surfaces be developed as an option for solid surface plasma facing components?

Liquid surface PFCs avoid the majority of the deleterious effects of alpha and neutron irradiation, erosion, and off-normal events. There are no thermal stresses in a liquid so cyclic fatigue and creep are not an issue. However, the peak operating temperature of a liquid surface is limited by evaporation from the liquid surface. The precise temperature limit also depends on the transport of the evaporated material to the plasma, which is uncertain. In general, the expected temperature limits are below those desired for the highest thermal efficiency for lithium, but could be acceptable for tin and gallium (with further assessment). For the continuing development of the liquid surface PFCs, the following requirements will need to be addressed.

- Magnetohydrodynamics (MHD) modeling and material transport at high Hartmann, Reynolds, and Magnetic Interaction numbers with fusion relevant fields, configuration and spatial and temporal gradients, will be required.
- Creative engineering of practical devices for injecting, controlling and removing liquid material will be needed.
- Laboratory testing of liquid surface PFCs in relevant magnetic conditions with heat and particle fluxes for the demonstration of operation effects, including wetting, chemical effects, and corrosive properties, should continue.

- Liquid surface PFC operation in a tokamak environment will be needed for the demonstration of heat removal, safety and diagnostics, and the coupling with physics performance.
- Innovative design for high-thermal performance liquid surface option for DEMO should be investigated.

INTERNAL COMPONENTS

Functional internal components (antennas, sensors, mirrors, control coils, etc.) must meet the criteria of other plasma facing components in terms of resistance to high heat and particle fluxes with acceptable levels of impurity production, while performing their functions. Most presently operating high-power toroidal fusion experiments are pulsed and have limited neutron fluxes (only those resulting from D-D operation). For these experiments, heat loads are handled inertially, and particle deposition on components can either be: tolerated (as with antennas or limiters), limited by the use of shadowing or shutters, or periodically removed or replaced (as required for mirrors, windows, or lenses) to continue functioning. Long pulse, nonnuclear experiments like Tore Supra add active cooling to all internal components, and development of these techniques has occurred progressively based on experience more than ~20 years. Newer nonnuclear, superconducting long pulse devices (LHD, KSTAR, EAST [operating] and W7-X, JT-60SA [under construction]) can be used as future test beds.

MEASUREMENTS

What improvements to diagnostic measurements are needed to validate the PWI models with precision?

Can effective front-end optics used for diagnostics be developed to survive the harsh neutron environment of a fusion plasma?

Burning plasma properties introduce new fundamental measurement limitations to some existing measurement techniques, as well as present an extremely hostile environment that challenges a range of diagnostics — particularly those requiring plasma facing, optical quality mirrors. Examples of both of these challenges are provided below.

In a burning plasma, relativistic effects (T_{e0} ~20keV) degrade the spatial resolution of widely utilized electron temperature measurement via electron cyclotron emission (ECE). Analysis by a scientific team led by the University of Texas has indicated an expected spatial resolution of 5 to 10cm in the core of ITER plasmas – almost an order of magnitude worse than currently available. In addition, edge pedestal temperature (a critical ITER performance parameter) measurements via ECE have a spatial resolution of ~4 cm at an edge pedestal temperature of 4keV, which does not meet ITER requirements and is likely inadequate to resolve the steep spatial gradient.

The above is an example where plasma parameters of a new regime degrade an existing measurement capability. The deterioration is not due to hardware limitations, but is **fundamental to the measurement technique itself.**

In contrast to such a fundamental limitation, there is also significant concern regarding a broad array of existing optical diagnostic techniques planned for ITER, such as Thomson scattering, charge exchange recombination spectroscopy, and motional Stark effect. Maintenance of the optical quality of mirrors, polarizers, windows, etc., located close to a burning plasma environment represents a significant, and perhaps overwhelming, challenge to overcome. In particular, redeposition and erosion combined with long plasma exposures present a forbidding challenge. In addition, the bremsstrahlung background emission will be far greater than observed in current devices and ultimately limit signal to noise. It seems highly likely that the availability and accuracy of such visible diagnostics will be severely compromised in the ITER environment in comparison with existing devices. This is especially the case during long pulse (~1000s) exposure to high heat and neutron loads during high-performance operation. Optical-based diagnostics requiring plasma facing mirrors will be extremely challenging for ITER (area of international concern) and effectively untenable on a DEMO.

The variety of challenges posed by a burning plasma environment to measurement capabilities must be urgently addressed. A primary goal for ITER is to understand the physics of burning plasmas. This will require a wide range of detailed measurements – ideally *superior* to current measurement capabilities. It is critical that immediate work begin to both identify and resolve the "gaps" in measurement capability so that ITER can prepare us for future burning plasma devices such as CTF and DEMO.

RF ANTENNAS AND LAUNCHERS

How can the predictive capability of plasma edge models, including material interaction, be enhanced?

Can these enhanced models incorporate the formation of radiofrequency sheaths produced by radiofrequency waves transiting between the radiofrequency antennas and absorption in the core plasma?

Can material with greater toughness and improved thermal conductivity be developed with specific materials, coatings, and heat sinks suitable for high-power radiofrequency components for the fusion environment, resulting in the handling of higher power density radiofrequency components that are robust and tolerant of plasma conditions?

Can innovative concepts be developed that move sensitive front-end components far from the plasma edge?

Many of the issues concerning the interaction of the near field of the antenna with the plasma in the SOL are not well understood. There is a need to predict, measure, adjust for, and modify this edge environment since it is the region through which the power is coupled. The formation of the radiofrequency plasma sheath can lead to focused particle and energy fluxes on the antenna and on surfaces intersected by the magnetic field lines connected to the antenna, with a dependence on the antenna structure and phasing. The resulting hot spot formation and enhanced local erosion serve as local impurity sources and can degrade core confinement. The parasitic radiofrequency losses in this region include edge modes (shear Alfvén and cavity modes), parametric instabilities, and nonlinear wave-particle interactions.

Integration of fully 3-D heating codes with realistic antenna and confinement device geometry is needed to understand the coupling of radiofrequency power from the antenna to the core, as well as for fundamental understanding of factors that will limit antenna operation, such as ELMs, parasitic losses, breakdown, and nuclear materials issues. To address these challenges, we propose an integrated effort that includes fundamental improvements in modeling coupled with experimental validation of the models on both dedicated test stands and confinement devices.

Other challenges concern the antenna straps, Faraday shield or front grill of launchers and additional nearby structure and surfaces. Radiofrequency breakdown/arcing is one of the main power limiting issues with operating the antenna in a plasma environment and is poorly understood. The anticipated large outer gap on DEMO (and ITER) and the resulting impact on loading will likely push operating voltages to at least as high as the ITER design limit of 45 kV. The interaction of the antenna surfaces with the plasma (including ELMs) and the role of the resulting particles and local gas load on breakdown and loading are issues that need improved understanding.

The antenna structure and Faraday shield will likely be constructed from layered or coated materials. Launcher materials require good conductivity and high heat resistance. The behavior of these structures in a nuclear environment and the survivability in long-term operations is a concern. The exposed antenna or launcher surfaces must be resistive to high heat (1-10 MW/m^2) and neutron fluxes with acceptable levels of impurity production. Launcher windows and antenna insulators must retain their function in high-radiation environments. Also, issues that result from operating at ~600 °C, as required by DEMO, need to be addressed.

For remotely maintained D-T experiments, the performance criteria for radiofrequency antennas and launchers become more demanding, as replacement or repair becomes complicated. Shadowing components may be required in some cases where extensive particle deposition can impede function.

In present-day ECH launchers, a final mirror is used to direct the microwave power toroidally and poloidally to achieve optimal coupling to the plasma. The final mirror must have a highly reflecting surface to reduce the heat load on the mirror itself. The obvious concern is that this surface will not last for many years when exposed to the burning plasma.

INTERNAL COILS

Can the severity of off-normal events be lessened or eliminated by using 3-D internal coil systems?

Can insulating and conductive materials be developed with specific materials, coatings, and heat sinks suitable for internal coil components for the fusion environment, resulting in components that are robust and tolerant of plasma conditions?

There are fundamental gaps in our understanding of internal control coils in future confinement devices as fusion research proceeds toward DEMO. Reliable and predictable cancellation of error fields, suppression of resistive wall modes, RMPs, and ELMs, and control of vertical stability cannot be achieved unless these gaps are closed. Integration of fully 3-D MHD codes with realistic control coil geometry is needed to understand how the coils affect the plasma and how the different coil sets interact with each other. Three-dimensional fields could have an impact on the SOL and possibly the divertor, and could result in toroidally asymmetric heat and particle loads on the PFCs. This could be beneficial by spreading the heat flux, or deleterious, by increasing the peak heat flux. The impact of these effects has not been considered for ITER or DEMO. To address these gaps, we propose an integrated effort that includes fundamental improvement in modeling coupled with experimental validation of the models on confinement devices. Basic research needs to be performed on the development of insulators and conductors that can perform in the high heat and neutron environment of a D-T DEMO. Studies need to be performed on appropriate methods for extracting the heat from the coils generated by resistive losses and nuclear heating.

Research Thrusts

The following table summarizes the research thrusts that will address the research requirements described above and bridge the knowledge gap on the path to DEMO.

Requirement	**Research Thrust to Address Requirement**
Diagnostic investments for edge characterization	1 and 9
Dedicated experimental time for edge characterization	1, 5, 9, and 10
Improved models and code components for edge region	9 and 10
Innovative divertor concepts and testing	9 and 11
Transient impact on plasma facing components	2, 6, and 10
Innovative design of solid surface PFCs	10 and 11
Testing necessary to validate codes, improved physics models, and new designs	10 and 11
Liquid surface development	11
Improved diagnostic parts in edge	1, 2, 9, 10, and 11
Antenna and launcher development	10
Internal coils	2 and 5
Integrated demonstration of taming plasma materials interactions	12

TAMING THE PLASMA-MATERIAL INTERFACE
THEME MEMBERS

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RESEARCH THRUST COORDINATORS

Thrust 9–Unfold the physics of boundary layer plasmas.
RAJESH MAINGI, Oak Ridge National Laboratory
TOM ROGNLIEN, Lawrence Livermore National Laboratory
Thrust 10–Decode and advance the science and technology of plasma-surface interactions.
DON HILLIS, Oak Ridge National Laboratory
Thrust 11–Improve power handling through engineering innovation.
DENNIS YOUCHISON, Sandia National Laboratories
Thrust 12–Demonstrate an integrated solution for plasma-material interfaces compatible with an optimized core plasma.
TONY LEONARD, General Atomics

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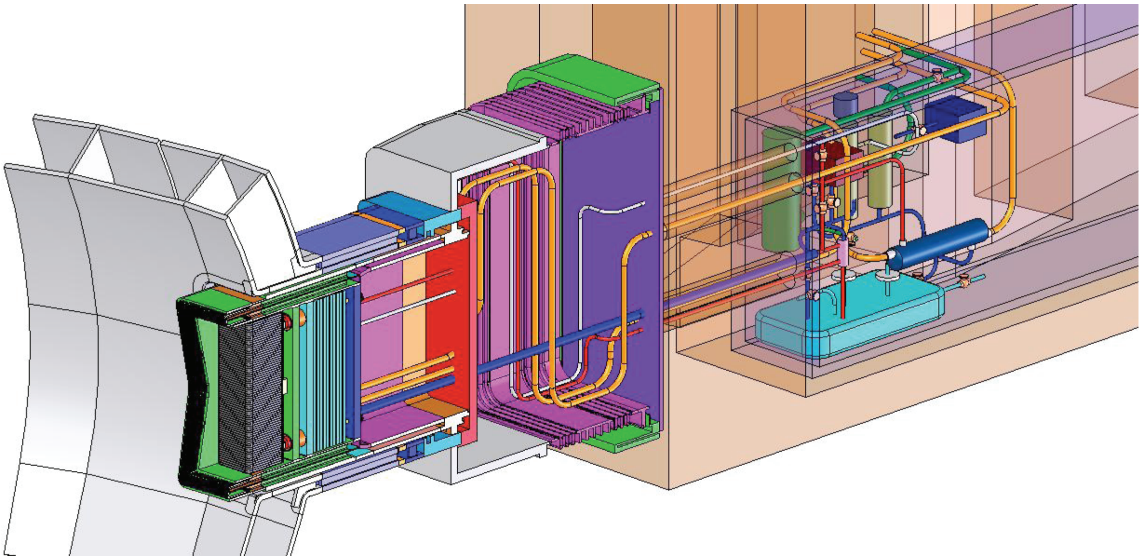
ROSTOM DAGAZIAN, Office of Fusion Energy Sciences, U.S. Department of Energy

THEME 4:
HARNESSING FUSION POWER



ON PREVIOUS PAGE
Design of an ITER Test Blanket Module, shown installed.

THEME 4: HARNESSING FUSION POWER

Introduction

SCOPE AND FOCUS

The Harnessing Fusion Power Theme addresses a set of science and technology challenges associated with designing and operating a fusion reactor. The issues encompassed in this Theme must be resolved to realize the ultimate benefits of fusion power; more urgently, their resolution lays a critical foundation for the construction of a demonstration fusion power plant (DEMO), which in the US is considered to be the last step before commercialization. The key elements included in this Theme are illustrated in Figure 1.

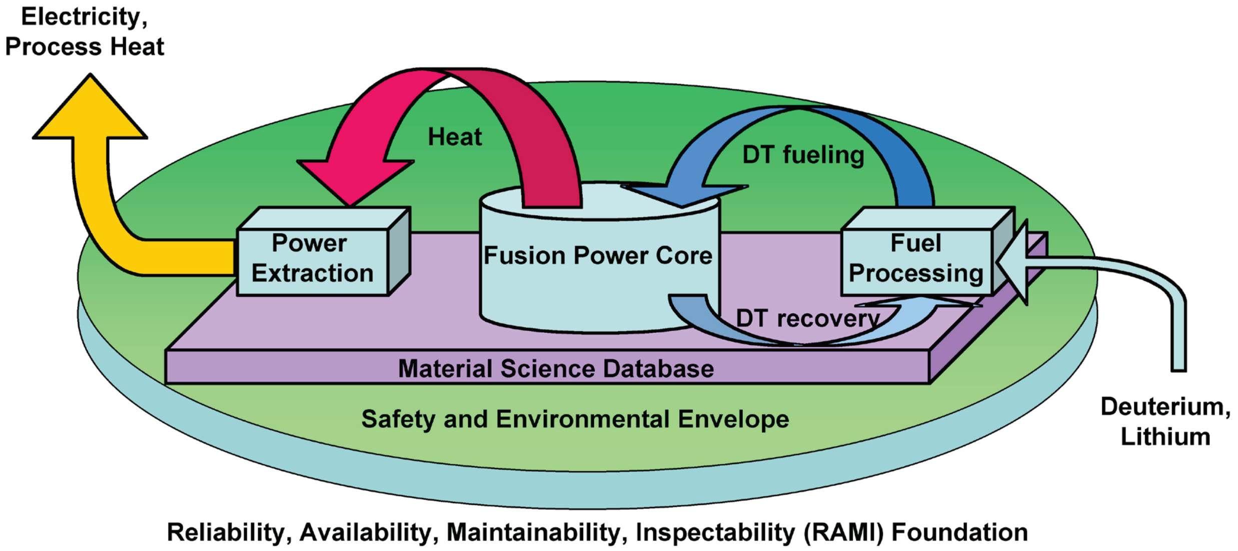


Figure 1. The Harnessing Fusion Power Theme was organized into five topics (Panels): Fusion Fuel Cycle, Power Extraction, Materials Science, Safety and Environment, and RAMI.

Fusion Fuel Cycle: Fusion power plants, including DEMO, will burn deuterium and tritium (D-T), producing 3.5 MeV alpha particles (helium) and 14 MeV neutrons. Deuterium occurs naturally in vast quantities, but tritium is radioactive with a short half life and is not naturally available in quantities needed to fuel fusion plants. Fortunately, tritium can be produced by capturing fusion neutrons in lithium. Consequently, a lithium-containing tritium breeding blanket will be placed outside the thin wall that surrounds the plasma. We must determine how to manage the flow of tritium throughout the entire plant, including plasma fueling, breeding, recovery and containment.

Power Extraction: About 20% of the fusion power is absorbed as surface heat by components exposed to the plasma (first wall and divertors), while ~80% is deposited volumetrically by neutrons in the breeding blanket. These components must reliably operate at a high enough temperature (e.g., > 600°C) to efficiently produce electricity or hydrogen. *We must develop the fusion chamber components that can achieve this capability.*

Materials Science: The high-energy neutrons can displace atoms and produce helium and hydrogen by transmutations in the materials, degrading their properties and performance. Materials used in the plasma chamber structures, tritium breeding blankets, plasma diagnostics, and heating systems will suffer damage. *We must understand the basic materials science phenomena of radiation damage and qualify components for use in fusion power plants.*

Safety and Environment: Fusion energy has the potential to be environmentally attractive. Through careful choice of materials, the induced radioactivity lifetimes can be very short compared to those from fission reactors. All fusion materials should be recyclable to minimize the radwaste burden for future generations. Because of these and other inherent safety features, studies of possible fusion power plants indicate they will be safe and should have no need for a public evacuation plan. *We must convincingly demonstrate the safety and environmental potential of fusion.*

Reliability, Availability, Maintainability, and Inspectability (RAMI): DEMO must demonstrate a high enough availability for power producers to build a commercial fusion plant. Power producers cannot expect an ultimate fusion power plant availability of 80% (or more) if DEMO cannot demonstrate a 50% or higher availability. Achieving this DEMO availability goal will require reliability in component design, design integration for RAMI, high maintainability, and systems to monitor and inspect components. *We must develop and qualify methods and capabilities needed to achieve RAMI objectives.*

HIGHLIGHTS OF PROGRESS

Although there are large gaps in the knowledge required prior to DEMO, significant progress has been achieved during the last 10-15 years in areas related to the harnessing of fusion power, despite limited funding. Examples of such progress are highlighted below for the different panel areas.

Fusion Fuel Cycle

- The Tritium Systems Test Assembly (TSTA) was designed, built and operated to demonstrate the fusion fuel cycle. This experiment, together with a number of other facilities, led to the successful D-T campaign on TFTR. These programs identified and solved many fusion fuel processing issues at a scale many orders of magnitude less than DEMO.
- The Mound tritium facilities and other facilities showed that large amounts of tritium can be successfully handled without harm to workers, the public or the environment.
- The US has been and is currently involved with the ITER fuel cycle. This experience has been valuable in preparing for DEMO.

Power Extraction

- Significant advance in predictive capabilities in key areas including, for example: coupling of CAD models with neutron and photon transport codes; developing 2-D and 3-D liquid metal magnetohydrodynamics (MHD) numerical methods and simulation tools coupled with heat transfer simulations capable of exploring very complex liquid metal blanket flow physics; dynamic models of tritium fuel cycle behavior (a key need to assess tritium self sufficiency); and MELCOR safety analysis code for power extraction components.

- Conception and initial development of the Dual Coolant Lead Lithium (DCLL) blanket concept, providing a potential pathway to high outlet temperature and high efficiency power conversion with current generation reduced activation materials. The DCLL is a significant driver of current research and international collaborations on silicon carbide (SiC) development and irradiation experiments, MHD experiments and simulations of flow control and heat transfer, and tritium behavior in lead lithium alloys.

- Models and experimental validation of tritium release from and thermal conductivity of glass-like ceramic breeder particle beds.

- First in-depth joint technology and physics investigation of the behavior of innovative liquid surface wall and divertor concepts. These were done as part of research programs, ALPS and APEX, that led to a much deeper understanding of issues and designs, as well as continuing experiments with lithium surfaces in confinement devices in the US and abroad.

- Development of conceptual designs of attractive magnetic fusion energy power plants, including advanced divertor and blanket concepts for high-efficiency power generation (helping identify key R&D).

- Initial designs and analysis of ITER test blanket module experiments for both the lead lithium and ceramic breeder blanket systems to fully utilize the first long pulse fusion environment for investigating power extraction component behavior and phenomena.

Materials Science

- Advances in the development of reduced-activation ferritic-martensitic steels with good thermal conductivity and thermal stress parameter, resistance to swelling at high dose, and with well-developed manufacturing technology for nuclear applications.

- Addressing key scientific questions concerning nanocomposited ferritic alloys. These alloys possess excellent high-temperature strength and show significant promise for mitigating the damaging effects of neutron irradiation, including the potential to tolerate high levels of transmutation-produced gases such as helium and hydrogen.

- Exploring the application of reinforced silicon carbide materials. While only a relatively small fraction of the fusion materials science research portfolio, the effort on fiber reinforced silicon carbide composites has demonstrated tremendous potential. These truly low-activation materials have both structural and functional applications in fusion power systems.

Safety and Environment

- Development and application of the Fusion Safety Standard for a consistent, integrated approach demonstrating the inherently safe and environmentally beneficial potential of fusion power.

- Conception, enhancement, and verification of nuclear safety simulation tools that incorporate the unique and challenging phenomena present in fusion power systems, allowing execution of the Fusion Safety Standard, for example, in the licensing process for ITER.

- Life cycle strategy development for fusion plant materials disposition that are under consideration by international parties. Recycling and restoration are key to the economic compatibility of fusion with environmental stewardship.

Reliability, Availability, Maintainability and Inspectability

- In the 1970s, the Electric Power Research Institute (EPRI) and other institutions determined that existing RAMI tools and risk assessment tools would apply to fusion.

- In the 1980s, safety analysts examined faults of fusion components to identify failure modes and hazards. Analysts started to create "generic" data sets to use for safety and reliability assessments of fusion designs. The Next European Torus RAMI study was the most comprehensive study of that time.

- In the 1990s, analysts collected actual reliability data from other facilities (e.g., accelerators, fission plants) to apply mainly to fusion experiment safety assessments. Some existing tokamaks (e.g., JET and DIII-D) accumulated enough operating hours late in that decade to yield statistically significant reliability data values from their operating experiences. The EU blanket reliability study was one of the most comprehensive studies of that time.

- In the 2000s, tokamak and tritium facility reliability, and to a lesser extent maintainability, data analyses have been performed on fusion systems and components with highest relevance to future tokamaks. These data analyses support ITER design.

- In the 2010s, plans are being made to collect detailed RAMI data from pertinent ITER systems to apply to next-step machines.

Research Requirements

FUSION FUEL CYCLE

The top level objective for the Fusion Fuel Cycle topic is to learn and test how to manage the flow of tritium throughout the entire plant, including breeding and recovery.

Requirements for DEMO

The fusion fuel cycle experience base for DEMO includes TSTA and other facilities such as the Savannah River Site. TSTA and related facilities are considered the present state-of-the-art experience for fusion fuel processing. But this experience will fall short of what is needed for ITER and DEMO. This progression from TSTA through ITER and to DEMO is summarized as follows:

Parameter	**State-of-the-art**	**Need for ITER**	**Need for DEMO**
Flowrate	6 liters/min	120 liters/min	500 liters/min?
Time required	24 hr	1 hr	1 hr
Tritium inventory	100 gm	4000 gm	6000 gm
Duty cycle	15%	5%	50%
Power	Designed for 1000 MW power plant	500 MW	2000 MW
Tritium breeding requirement	None	1.4 kg tritium burned per year (none bred in original phase)	Must breed all tritium

Scaleup of the fuel cycle for DEMO will require new technologies and challenge existing technologies. This scaleup will underlie much of the fuel cycle needs.

Fuel Cycle Approach

Considering the fuel cycle in the context of DEMO (substantial scaleup), the following aspects will be challenging: 1) fusion fuel processing, 2) vacuum and fueling, 3) containing and handling tritium, 4) performing tritium accountability and nuclear facility operations, 5) breeding tritium, 6) extracting tritium from the breeding system, and 7) in-vessel tritium characterization, recovery and handling. Each area will be described along with the state-of-the-art and gaps between the latter and what is needed.

FUSION FUEL PROCESSING (Need 1)

Description: The fusion fuel cycle is composed of the following sub-systems: fuel cleanup (for ITER Tokamak Exhaust Processing), isotope separation, tritium storage and delivery, water detritiation, tritium pumping, effluent detritiation, gas analysis, and process control.

State-of-the-art: The state-of-the-art for each of these systems was developed at TSTA, JAERI, FzK, JET, SRNL, TFTR, Chalk River and other facilities. These systems were typically tested at 1/20th the scale of ITER (or less). ITER will be a major technological challenge and much will be learned from it. The ITER tritium systems will largely be a production system with little opportunity for experimentation. DEMO will require higher throughput (4 x ITER) and a higher duty factor (10 x ITER)

Gaps: Due primarily to scaleup, all DEMO sub-systems will require improvements including: better technology, tritium inventory minimization, accuracy improvement, increased throughput, avoidance and/or handling tritiated water, improved duty cycle and design and diagnosis tools.

VACUUM AND FUELING (NEED 2)

Description: The vacuum and fueling systems are composed of the key sub-systems: torus vacuum pumps, roughing pumps, gas puffing, pellet fueling, disruption mitigation and ELM pacing (an ELM is an edge localized mode – an instabiliy affecting the plasma near the vessel wall). Torus vacuum pumping must maintain low divertor pressure (~10 Pa) while removing helium ash that will be generated by the fusion burn. The fueling system must provide D-T fuel to the burning plasma and gas to the scrape-off layer (SOL) and divertor to minimize impurity generation and sweep impurities to the divertor. Furthermore, systems must be provided for massive gas injection or other systems for disruption mitigation and rapid small pellets for ELM pacing.

State-of-the-art: The pumping system for ITER consists of eight cryosorption pumps that are regenerated every five minutes in a cyclic fashion. These pumps are backed by tritium compatible roughing pumps (still under development). Frequent regeneration will be challenging. The pellet fueling system for ITER will be become the state-of-the art, but the DEMO requirement will be more demanding (4 x ITER on flowrate and 10 x on duty cycle). Pellet penetration requirement may need to be increased (4 x ITER on flowrate and 10 x on duty cycle). The state-of-the-art for disruption mitigation is gas jets. ELM mitigation with pellet pacing is not well developed. The next few years will hopefully answer the question of whether this could be employed. The requirements for disruption and ELM mitigation in DEMO are completely unknown at this point. These requirements could have a significant effect on the fueling and pumping systems as well as the overall fuel cycle design.

Gaps: DEMO will require improved vacuum systems such as roughing pumps and cryopumps. Pumps that separate species have advantages. Gaps for the fueling systems are to be determined based on unknowns for DEMO such as fueling penetration requirements, feed rate, tokamak/not tokamak, etc. Also, the gaps for DEMO disruption mitigation and ELM pacing are presently to be determined.

CONTAINING AND HANDLING TRITIUM (NEED 3)

Description: Tritium is hazardous to workers, the public and the environment. To mitigate this hazard it must be properly contained and handled. Systems for this consist of: primary, secondary and tertiary containment; permeation barriers; occupational and environmental monitoring; maintenance systems; waste handling, characterization processing and disposal; decontamination and decommissioning; and personnel protection equipment.

State-of-the-art: Experience at recent tritium facilities. ITER will be challenged in this area and DEMO will be an even greater challenge with high-temperature operation, useful power extraction and higher duty factor.

Gaps: Control of tritium through nontraditional tritium-handling equipment (heat exchangers; large, high-temperature components; long high-temperature pipe runs) will require significant attention. Room processing systems will be challenging. Permeation barriers would help, but development has not been successful, and the barrier factor is diminished under irradiation.

PERFORMING TRITIUM ACCOUNTABILITY AND NUCLEAR FACILITY OPERATIONS (NEED 4)

Description: A facility with significant amounts of tritium must operate with a methodology which ensures that the facility's tritium is not a practical threat to workers, the public and the environment, and that it has not been diverted from the facility. This area consists of 1) tritium accountability measurement techniques, 2) tritium accountability methodology and procedures, 3) nonproliferation approaches, 4) systems and approaches to ensure worker and public safety (authorization basis), 5) tritium transportation technology and approaches, 6) waste repository, and 7) tritium supply.

State-of-the-art: Accountability measurements are performed by in-bed calorimeters or off-line P-V-T methods. Processing times are long and accuracies are limited. Accountability methods rely on periodic reconciliation between "book" inventory and "physical" inventory. Proliferation/diversion is dictated by an "attractiveness level" as defined in the US by DOE Orders (tritium is considered less attractive than special nuclear materials, but still requires safeguards such as "Gates, Guards, and Guns" with access restrictions). The authorization basis is defined in the US by DOE, NRC, Fusion Safety Code and ITER requirements. Risk-based assessments are used for calculating dose to the public. "Agreements" are made between contractors and DOE for chronic emission limits from facilities. Waste facilities are covered in the Safety section.

Gaps: Presently under consideration for the ITER torus are tritium accountability measurements based on "inventory-by-difference" and unavoidable errors predicted to propagate quickly. Direct methods of estimating tritium inventory need to be developed. With increased potential hazard, DEMO will require improved accountability methods. The DEMO authorization basis will require balancing cost and safety. An important strategy will be demonstrating methodology effectiveness. Tritium waste disposal must be addressed early to ensure that facilities can effectively operate and shutdown. Integration with Safety will be important.

BREEDING TRITIUM (NEED 5)

Description: To date, fusion operations requiring tritium have had tritium supplied from non-fusion sources (around 20 kg total available). For success, however, DEMO will have to breed essentially all of its required tritium. Tritium breeding can be broken down in the following areas: 1) blanket materials and configurations, 2) blanket structural materials, 3) blanket operations and control, 4) blanket maintenance and disposal and 5) blanket diagnostics.

State-of-the-art: A number of solid and liquid breeder concepts exist. Currently the primary development of both liquid and solid concepts is in the EU and Japan.

Gaps: No tritium breeding proof-of-principle has been performed, although any major tritium-using facility other than ITER will have to breed its own tritium.

EXTRACTING TRITIUM FROM THE BREEDING SYSTEM (NEED 6)

Description: To be useful, bred tritium must be extracted from the breeding material. Less soluble materials make it easier to extract the tritium but may suffer from containment issues and vice versa. It is envisioned that solid breeders will have tritium extracted from the breeder materials by sweeping helium through the breeder. Limited fundamental experiments have been performed. Liquid breeders will flow the breeder and the tritium away from the torus, and a processing vessel will extract the tritium. Some experiments of the latter type have been performed.

State-of-the-art: Data-to-date suggest that tritium recovery from the breeding material with acceptable tritium inventory is feasible, but this has not been performed in an integrated fashion with tritium containment. Only preliminary tests have been performed.

Gaps: Need to select and test the tritium extraction methods, and demonstrate that tritium can be reduced to levels that will not challenge containment systems. Need to include extraction from beryllium. Testing in concert with 14 MeV neutrons, high burn up and high flux is needed.

IN-VESSEL TRITIUM CHARACTERIZATION, RECOVERY AND HANDLING (NEED 7)

Description: Tritium deposition inside the torus is expected. Calculations have shown that tritium can rapidly accumulate in certain ITER conditions. Methods are needed to characterize in-vessel tritium. Also needed are methods to recover in-vessel tritium and handle in-vessel components that have deposited tritium.

State-of-the-Art: Experience has been gained with tritium experiments on TFTR and JET. Stand-alone experiments have shown that tritium buildup on carbon machines is significant and less so on tungsten machines. Higher first-wall temperatures will help.

Gaps: Presently there is no tungsten plasma facing component testing data in a DEMO-like nuclear environment. There is a need to test the tritium hold-up on the tungsten divertor and first wall under DEMO-relevant conditions. There is also a need to test and develop knowledge to increase the burn-up fraction and recycling coefficient under relevant toroidal plasma conditions.

POWER EXTRACTION

Power extraction is a fundamental challenge for an attractive fusion energy source. The fusion plasma deposits significant energy in the form of energetic particles and X-rays on the surfaces of the components that surround the plasma. In addition, neutrons born from the fusion reaction carry their energy deeper into the components, resulting in strong volumetric heating. The goal of fusion power extraction is then to capture this energy, transport it away from the burning plasma, and convert it efficiently to electricity or some other useful form such as hydrogen or process heat. Conversion of this energy at high efficiencies requires that the coolant streams performing the power extraction are at high temperature.

The element lithium also must capture the same neutrons carrying 80% of the fusion power to generate (or *breed*) tritium fuel needed to resupply the plasma. The combined system that captures these neutrons for both breeding and energy harvesting is called the *blanket,* and it occupies about 90% of the surface area surrounding the plasma. It has an integrated *first wall* facing the plasma that also captures a portion of the surface energy flux from the plasma. High-temperature power extraction must be accomplished using strategies, components and materials that do not damage the potential to continuously breed the tritium fuel. For instance, using thick structures and plasma facing surfaces to increase strength and absorb energy is not possible because of parasitic neutron absorption and the resultant decrease in tritium breeding potential.

The remaining portion of the surface heating is typically concentrated on a second power extraction component called the *divertor,* designed to take very high surface power loads. For the divertor (and to a lesser degree the first wall), power extraction issues are integrally connected with those regarding plasma surface interactions, a topic specifically treated in Theme 3.

The blanket and the divertor have an additional radiation shield behind them to prevent any remaining radiation from damaging the vacuum vessel and superconducting magnets. The vacuum vessel is a large structure that contains the fusion plasma, allows for good vacuum conditions to be created, and serves as a safety confinement barrier for radioactive materials. The shield captures only a small amount of the total energy, which is typically removed with a low temperature coolant and discarded as waste heat.

All of these systems are connected to systems that deliver new coolant, accept the returning heated coolant, and couple that stream to some thermodynamic power cycle. Coolant circulation and power conversion systems must be both highly safe and reliable, as they communicate between the plasma and the balance of the plant, transporting energy, and possibly tritium and radioactive impurities that must be strictly controlled.

All these power extraction components are inside the vacuum vessel and in immediate proximity to the plasma. This has strong implications on design, materials, maintenance and reliability requirements for such components. Power extraction components:

- Cannot have any leaks without spoiling the vacuum.
- Must tolerate significant heat flux and plasma erosion of surfaces, including off-normal events like plasma disruptions that produce severe surface energy pulses.
- Must be replaceable in reasonably short times.
- Are damaged by fusion neutrons and plasma particles, and so have evolving material properties and a limited lifetime.
- Have complicated geometries to conform to the shape (toroidal in a tokamak) of the plasma and accommodate many penetrations for plasma fueling, heating, and instrumentation equipment.
- Are electromagnetically coupled to the plasma in complicated ways and so must be designed for compatibility with plasma operations, including off-normal events like plasma disruptions that induce severe electromagnetic forces.

These significant constraints and interrelationships make fusion power extraction a challenge.

Compelling Scientific Issues

The goal of fusion power extraction research is to establish the scientific underpinnings and technological development needed to efficiently, safely, and reliably capture and transport fusion energy while remaining compatible with tritium breeding and plasma operation. The scientific issues encountered are related mainly to understanding the:

- Thermal and fluid dynamics behavior of liquid metal and gas coolants.
- Thermal and mechanical behavior of the materials and structures.
- Generation and transport of tritium, corrosion and undesirable radioactive impurities.

- Neutron and other radiation transport, deposition and secondary nuclear reactions.
- Effect of material changes and realistic fabrication processes on the component behavior and design.
- Techniques for measurement of phenomena and compatible instrumentation.
- Synergistic phenomena driven by the unique combination of operating conditions.
- The key life-limiting and failure mechanisms, and how to identify and extend them.
- The interactions among plasma operation, tritium fuel cycle, material, safety and secondary electrical generating systems.

Power extraction issues differ substantially from other energy sources, including fission, due to the extreme conditions, multiple competing requirements, and the unique multi-physics environment in which fusion power extraction components and materials must function. "Traditional" approaches and materials for high heat flux and nuclear heat removal are usually not applicable in fusion because of these unique environmental factors and requirements, eliminating, for instance, the use of many materials due to irradiation damage and activation concerns. Specialized combinations of materials including high-temperature coolants, tritium breeders, neutron resistant structures, neutron multipliers or reflectors, electrical and thermal insulators, etc., are necessary, leading to complex components with many interfaces. Unique phenomena such as magnetohydrodynamic interactions of liquid metal coolants with the magnetic field, or radiation assisted corrosion and tritium permeation, have a strong impact on the ultimate performance of the power extraction components. The scientific understanding required to advance fusion nuclear technology to a point sufficient to assure the feasibility and competitiveness of fusion energy, or perform a practical and licensable design for a DEMO, must involve a significant database obtained in relevant conditions.

Gaps and Research Requirements

While significant progress has been made in understanding power extraction phenomena, new issues and even more complicated behavior have been uncovered requiring further study and more in-depth characterization. Knowledge in many areas and scientific disciplines is still required. The current view of these knowledge gaps are summarized below:

- A sufficiently complete understanding of component spatial and temporal temperature variations, thermomechanical effects, coolant flow, and transport phenomena at fusion relevant parameters.
- The impact of and techniques for control of coolant chemistry and impurities for fusion-relevant coolants and conditions.
- Knowledge about the occurrence, severity and consequences of synergistic phenomena resulting from multi-field, multi-material, or multi-function effects.
- Sufficient techniques to fabricate, maintain, and diagnose experiments, modules, and components with fusion-relevant materials at fusion-relevant conditions.

- Integration of knowledge base into models capable to predict component behavior beyond parameters of the experimental database and acceptable to design and license components for DEMO.
- Adequate knowledge to evaluate alternatives to mainline concepts and materials for blankets, first walls, and divertors with either different feasibility issues, or potential for increased margin or performance.
- Understanding how multiple power extraction systems, components and processes should be most effectively integrated to maximize performance, safety, reliability, and economy.

Progress toward the resolution of these power extraction gaps will only occur through a complementary and well-integrated program of computational modeling and well instrumented experimental campaigns, at first in laboratory test facilities, and later in increasingly integrated, dedicated facilities. There is still significant fundamental, separate effect research that must be undertaken, particularly in understanding fluid flow, heat and mass transfer effects, and material science and chemistry issues. The capability to perform multiple-effect to fully integrated fusion environment experiments is also a key. A set of experimental test facilities that can access different combinations of fusion relevant parameter ranges in magnetic field, heating, temperature, neutron and plasma energy and intensity will be required to advance the technological readiness level for power extraction beyond the current level.

Access to a fully integrated fusion environment suitable for testing of integrated experiments will also be necessary. The significant interactions among plasma, fuel cycle, power extraction, materials, safety, and reliability have to be investigated in an integrated fashion. For instance, utilizing ITER as a test environment will allow the initial exploration of synergistic effects, especially for prompt thermo-mechanical and thermo-fluid flow phenomena, including all-important environmental conditions for fusion. A more intense integrated fusion environment with longer pulses and more intense neutron irradiation will also be required. We envision a dedicated facility to produce such conditions, possibly based on a driven fusion plasma neutron source. In such a facility, larger scale, multiple mockups of blanket, first wall, divertor, and shield components can be exposed to a prototypical environment for significant time and with significant radiation damage. Only through such experimentation can we gain both the full understanding and confidence in components necessary for the final step to a DEMO reactor.

To properly perform this experimental science campaign, measurement techniques, sensors, attachments, and data transmission and processing systems will be required for temperature, flow, strain, pressure, vibration, leak detection, irradiation characteristics and many others. These measurement technologies must be compatible with a wide range of temperatures, materials, plasma, irradiation, and electromagnetic conditions. They must be integrated into experiments and modules without disturbing the function of the component or the phenomena being studied. Significant effort is required to prepare for and execute these increasingly integrated and challenging experimental investigations.

Emphasis on modeling and expertise in high-performance computing must also be a key element of addressing power extraction gaps. Incorporating the knowledge base into models and simulation tools gives researchers the capability to predict component behavior beyond parameters of the experimental database. Development of integrated simulation of the multi-physical phenomena unique to fusion systems is an important requirement. This capability is critical in fusion where true DEMO reactor conditions will likely not be fully reached prior to DEMO itself. As is clear from ITER experience, a fusion DEMO will be a nuclear device that must be designed and fabricated using vigorously validated codes acceptable to nuclear standards and regulatory agencies.

MATERIALS SCIENCE IN THE FUSION ENVIRONMENT

Overcoming the challenges confronting first-wall, blanket, plasma facing and functional materials systems is as difficult and important for fusion energy generation as achieving a burning plasma. Fusion materials and structures must function in a uniquely hostile environment that includes a combination of high temperatures, reactive chemicals, time-dependent thermal and mechanical stresses, and intense neutron fluxes. Atomic displacement damage alone in a DEMO-type reactor will be equivalent to ejecting every atom from its lattice site up to 200 times. Displacement damage undergoes complex interactions with high concentrations of reactive and insoluble gases that lead to the degradation of a host of performance sustaining material properties; these include hardening, low-temperature embrittlement, phase instabilities, segregation, precipitation, irradiation creep, volumetric swelling, and high-temperature helium embrittlement. Plasma facing components must also withstand heat fluxes comparable to those experienced by rocket nozzles. While key fusion structures are subject to a wide variety of poorly understood failure paths, they must clearly demonstrate high safety margins over long lifetimes. Indeed, the unprecedented demands placed on materials from thermo-mechanical loading alone are a feasibility issue, even without the severe effects of radiation damage.

Research Gaps

The materials panel considered the research gaps presented in the ReNeW resource document titled "Priorities, Gaps and Opportunities: Towards A Long-Range Strategic Plan For Magnetic Fusion Energy" that was published in October 2007. After careful consideration the panel decided to reformulate the gap statements to provide a more comprehensive assessment of the numerous challenges facing the many materials needed to construct a prototypical fusion power system. The following seven gaps capture the essential elements of what is missing from the current fusion materials research portfolio.

Gap: *The overarching scientific challenge facing successful development of a viable fusion power system is acquiring a firm scientific understanding and devising mitigation strategies for the deleterious microstructural evolution and property changes that occur to materials in the fusion environment.*

Although considerable progress has been made exploring the resistance of structural materials to neutron irradiation in fission reactors (to damage levels on the order of ~30 dpa), the current knowledge base for reduced-activation structural materials exposed to fusion-relevant irradiation conditions is almost nonexistent. In a fusion reactor structural (and other) materials will be exposed to simultaneously high-heat fluxes (up to 15 MW/m^2), high-neutron fluence (up to 20 $MW\text{-}y/m^2$), high concentrations of transmutation produced gases (~2000 appm helium and ~8000 appm hydrogen), and high time-varying thermal and mechanical stresses. The effects of helium on microstructural evolution have been investigated to near fusion-relevant levels in a limited set of materials systems in time-accelerated ion irradiation experiments, but these simulation experiments were not able to provide the bulk mechanical and physical property information needed by fusion power system designers. A unique aspect of the D-T fusion environment is the simultaneous production of displacement damage and very large quantities of gaseous transmutation products such as helium and hydrogen. Most of the hydrogen may diffuse out, but some may be trapped at bulk defects and interfaces, enhancing such phenomena as void nucleation and hydrogen-embrittlement. Due to its low solubility, helium precipitates into clusters or gas bubbles. At high temperatures, grain boundary helium bubbles grow and coalesce under stress, resulting in severe degradation of creep and fatigue properties. At lower temperatures, there is growing evidence that high helium synergistically interacts with displacement damage induced hardening, resulting in severe degradation of fracture toughness and intergranular fracture. At intermediate temperatures, He bubbles may serve as nucleation sites for growing voids, potentially leading to swelling and enhanced creep rates. A critical need is the capability to investigate the effects of neutron irradiation on bulk material properties while producing fusion-relevant levels of transmutation gases. The greatest need is for structural materials (e.g., ferritic steel for the blanket and W-alloys for the divertor), but also critical are other materials systems such as high-heat flux plasma facing components, breeding blanket materials, and a wide variety of functional materials essential for successful operation of a fusion power system.

Gap: *Development of high-performance alloys and ceramics, including large-scale fabrication and joining technologies, is needed.*

We anticipate that the current generation of reduced-activation structural materials may not provide adequate performance for a safe, environmentally sound and economically attractive fusion power system. There is a compelling need to evolve the next generation of candidate materials to increased levels of performance. We must seek breakthroughs in materials science and technology to discover a revolutionary class of high-performance materials for a fusion power system that fulfills safety, economic and environmental attractiveness goals. This can only be accomplished by implementing a full-scale alloy and ceramics development program. This includes the need to research large-scale fabrication and joining technologies for complex structures, including investigation of near-net-shape fabrication technologies that may provide substantially lower delivered costs for high-performance components. The most appropriate fabrication and joining techniques significantly depend on material selection, component geometry and expected service conditions. The traditional approach to developing the needed technologies relies heavily on trial- and-error approaches. A science-based program is needed to determine the applicability of recently developed technologies (for example, friction stir welding) for joining materials such as nanostructured ferritic alloys.

Gap: *Current understanding of strength-ductility/fracture toughness relationships is inadequate to simultaneously achieve high-strength and high-ductility and fracture toughness.*

The lower operating temperature limit for most fusion structural materials is determined by radiation hardening and embrittlement. At irradiation temperatures less than about 30% of the absolute melting point the defect structures produced by neutron damage significantly increase tensile strength, and decrease ductility and fracture toughness. Although the experimental parameters that cause radiation hardening and embrittlement are well known, the fundamental mechanisms responsible for loss of ductility are not fully understood. Improved multi-scale models coupled with appropriate experiments are needed to determine the causes of ductility loss in irradiated metals and suggest possible metallurgical solutions to ameliorate this effect.

Standard measurement methods of fracture toughness are not adequate for fusion power systems because the underlying principles of elastic-plastic fracture mechanics are violated for small specimens or in thin-walled structures with shallow cracks. A new method has been developed to provide a highly efficient means of acquiring and applying fracture toughness information, but further work is needed to verify its technical basis and account for embrittlement effects due to synergistic hardening and non-hardening mechanisms including helium and hydrogen.

Gap: *Predictive, physics-based, multi-scale models of materials behavior in the fusion environment are needed for development of advanced materials.*

The cost and time required to design, perform and examine materials from reactor irradiation experiments is high. Further, there is a combinatorial problem in that the broad range of materials, phenomena, irradiation variables and variable combinations makes a purely experimental approach intractable for fusion materials development. Robust computational models provide a means to reevaluate existing data, optimize the design and execution of new experiments, and interpret the results from those experiments. Physically based multi-scale models describing irradiation and mechanical damage processes for fusion applications are being developed, but numerous fundamental details remain to be fully resolved. Models of key properties that simultaneously span spatial and temporal scales ranging from the atomistic to the continuum and from sub-picosecond to years are urgently needed.

Gap: *Fundamental understanding of high-temperature deformation mechanisms is lacking.*

For fusion to be an economically attractive power source it must possess high thermodynamic efficiency. This increases the demand for structural materials capable of operation at high temperatures where materials can plastically deform at stresses well below the elastic limit. This phenomenon is known as thermal creep and is typically a concern for materials operating at temperatures greater than about half the absolute melting point. While considerable research has been performed on the mechanisms of thermal creep, only an elementary knowledge of the rate controlling processes is currently available. Present models tend to be semi-empirical in nature, do not fully capture the relevant physics associated with the deformation mechanisms operating at these temperatures, and generally have limited predictive capability, even for relatively simple metals and alloys.

The structures and components of fusion power systems will also experience cyclic (fatigue) loading. The interaction of fatigue and creep deformation is poorly understood. Existing creep-fatigue design rules are not based on the principles of materials science, but rather on experimental measurements for specific materials and conditions. The traditional approach to establish acceptable creep-fatigue operating limits relies on extensive testing, and large variations in these limits are found for various materials. Current ASME [American Society of Mechanical Engineers] creep-fatigue design rules are based entirely on empirical fits to experimental data. Hence, there is a compelling need to apply advanced experimental techniques and sophisticated computational methods for a better physical description of high-temperature deformation processes.

Gap: *The mechanisms controlling chemical compatibility of materials exposed to coolants and erosion of materials due to interaction with the plasma are poorly understood.*

For efficient power conversion, a high coolant exit temperature is desirable, which requires high-temperature structural materials. A potential limit on the upper use temperature is dictated by interaction of the structural material with the coolant. If the coolant/breeder material is not compatible with the structural material, its high temperature capability cannot be fully exploited.

Many variables affect the corrosion rate of a structural material in a particular coolant, including temperature, chemical composition of the structural material and the coolant, coolant flow velocity and velocity profile, coolant impurity concentration, and radiation. Most of the corrosion studies performed to date involve exposing a structural material in static coolant at a particular temperature, or a flowing coolant in a temperature gradient. Corrosion rate measurements are empirically correlated to coolant temperature and flow velocity. These correlations do not capture the fundamental physical mechanisms involved in the corrosion process and, therefore, are not useful for predicting material performance outside the range of the experimental measurements. A significant opportunity exists to improve the scientific understanding of corrosion processes through applying advanced analytical techniques and modeling methods such as in situ electrochemistry and computational thermodynamics codes.

Gap: *An integrated materials-structure development, design and testing approach to fusion systems, including facilities to determine the fundamental performance limits of materials and components in a fusion-relevant environment, is lacking.*

Materials and components for use in fusion power systems must ultimately be qualified to validate the design and demonstrate adequate safety margins. Fusion power systems must cope with time-dependent material properties in components with complex stress states, and long intended service lives. Existing thermo-mechanical property data and high-temperature design methodologies are not adequate to permit the design of a robust fusion power system. Development of fusion relevant design rules will require close integration between materials development activities and system design processes. Thus, fusion materials development necessitates the design of material systems and development of multifunctional structures concurrently. The traditional "function-oriented" material design approach will not be sufficient. Instead, a concurrent "materials-component-structure design" process must be implemented.

Research Needs

Critical research needs identified from the gap analysis are described below. The most urgent need is the capability to investigate the behavior of fusion materials under high levels of displacement damage while simultaneously producing helium and hydrogen. Consequently, a high-energy neutron source such as the International Fusion Materials Irradiation Facility is essential for developing and qualifying the full range of materials (first-wall, blanket, plasma facing components, functional materials, magnets, etc.) needed to construct a fusion power system.

There is also a compelling need for a suite of nonnuclear and nuclear facilities for testing materials, components and structures to investigate for potential synergistic effects not revealed in simpler single-variable experiments or limited multiple-variable studies. Such data is essential to refine and qualify predictive models of materials behavior, and provides essential reliability and failure rate data on materials, components, and structures to validate codes for designing intermediate step nuclear devices and a fusion power system. These facilities also provide a test bed to evaluate nondestructive inspection techniques and procedures as well as remote handing and maintenance technologies. Also included would be facilities for performing corrosion and compatibility studies, and characterizing the high-temperature mechanical properties of fusion structural materials.

Another critical need is to greatly expand and enhance the existing theory and modeling effort with the objective of developing experimentally validated multiphysics, multi-scale models that can predict the behavior, failure paths, and lifetimes of materials in the fusion environment.
Other key elements of a reinvigorated materials and engineering sciences program include:

Establishment of an alloy and ceramics development program to improve the performance of existing and near-term materials, while simultaneously pursuing breakthrough approaches to create materials with revolutionary performance characteristics (e.g., high strength, high toughness, high resistance to radiation damage and gases, exceptional thermo-physical properties, etc.). Couple this with a substantial effort on large-scale fabrication and joining technologies.

- An important crosscutting activity is the establishment of an integrated, concurrent materials-component-structure development, design and testing approach to fusion power systems. This activity begins with materials selection and ends with the qualified materials, components, structures and validated codes to describe the full range of behavior in the fusion environment.
- Extensive computational resources will be needed at all phases of fusion materials development to support model development. In particular, large-scale structural damage mechanics computational capability will be needed to guide and interpret data obtained from component-level test facilities.
- In addition to physical facilities, the need for human resources is particularly acute. A rough estimate of the workforce needed to carry out this program is on the order of ~60 FTEs at peak activity. Many specialized skills will be needed such as physical metallurgists, ceramists, electron microscopists, mechanical property specialists, fracture mechanics, materials evaluation specialists, corrosion scientists, nondestructive

evaluation specialists, theory and modeling experts and so on. The pool of available talent is shrinking rather than growing, which is a major concern.

SAFETY AND ENVIRONMENT

The FESAC Priorities Panel Report ("Greenwald report") confirms the potential for safe and environmentally sound fusion power through a sufficiently comprehensive development pathway. Fundamental knowledge of the underlying physical phenomena unique in fusion safety and environmental assessments is vital to ensuring public and worker protection via aspects identified in the Priorities Panel Report: plant licensing and commissioning, normal operation, off-normal events, postulated accident scenarios, and decommissioning. The broad areas of safety and environment gaps addressed in the Priorities Panel Report can be categorized as:

- Analysis tools for demonstrating compliance with the no-evacuation criteria.
- A database of materials properties and transport behavior in accident conditions suitable for benchmarking and validating the analysis tools.
- A protocol for developing codes and standards for fusion systems safety component qualification.
- An integrated strategy for activated materials management throughout the entire lifecycle of fusion facilities.

Harnessing fusion power for energy production must recognize safety and materials lifecycle management as fundamental attributes to all aspects of development. The following sections provide greater perspective on the importance of these two topics.

AN INTEGRATED SAFETY APPROACH

A critical element overlaying gap identification and prioritization is the importance of adequate safety integration into all levels of fusion facility design, including not only power plants but also test facilities such as IFMIF, FNSF [Fusion Nuclear Science Facility], etc. The importance of such integration is presently being demonstrated as the detailed design and construction of ITER proceeds. A basis of functional requirements established in the DOE Fusion Safety Standard was implemented into the ITER design and safety assessment. The approach has led to the early authorization of ITER's construction as a nuclear facility under the French regulatory authority. Developing a strategic framework and outlining the research needs for progress in fusion energy science is most appropriately performed in a fashion requiring safety integration. Thus, research gaps in the area of safety, particularly in identifying fundamental scientific needs, are effectively addressed in broad fashion when concept definitions mature and a prioritized development pathway is established.

Important lessons were learned during the various phases of ITER design with the integration of safety experts on the design team. The comprehensive role of safety in ITER includes:

- Establishing ITER safety criteria, starting with offsite release criteria and working down to individual systems. This step was required to effectively implement safety policy.

- Proposing integrated safety design approaches. For example, removing safety constraints from in-vessel components to the extent possible and evolving coordinated and comprehensive radiological confinement boundaries. Safety issues served as a common link among system design teams.
- Assisting system designers with safety assessments. The performance of each system in normal, off-normal, and accident situations is the responsibility of the designers.
- Assessing the design of systems with respect to their impact on other systems, so that safety experts focused on potential events could evaluate where one system affects another and system interconnections.
- Identifying potential safety issues early in the design, and performing safety research and development activities.
- Iterating on the functions above so that new design specifications or data were incorporated, problems with one system would be balanced elsewhere, and a coordinated confinement system design evolved as the various system source terms and their interactions with systems were better understood.

A key safety decision made early in the ITER project was to shift the safety burden away from plasma physics, plasma control, diagnostics, the divertor, first wall/blanket, and magnets to vessels, heat transport systems, and the tritium plant. By definition, ITER's physics and plasma facing components would be experimental, and an ITER objective was to maximize the flexibility of testing and experimentation during operation. This approach reduced the safety risk by removing all safety needs from the components and systems whose behaviors are least known. As a result, no plasma facing components served as part of the radiological control boundary, and wide allowances were preserved in case of their failure. The ITER safety design did, however, require limiting the in-vessel inventory of dust and tritium, thereby placing greater safety burden on the first confinement boundary.

One of the important design integration issues was defining the first confinement boundary. Choice of the vacuum vessel was a logical alternative that took advantage of preexisting fusion design approaches of high quality and high reliability vacuum vessels. The vacuum vessel can meet the low failure rate criteria with robust construction and double walls that confined tritium, neutron activated materials, and chemically toxic materials. Unfortunately, there are many vacuum vessel penetrations that are necessary to operate the tokamak. The integration challenge was determining the boundary perimeter for these penetrations. The designers required many vacuum vessel interfaces to the vacuum pumping system, the radiofrequency plasma heating systems, the fueling system, the diagnostics and their ports, penetrations for cooling system piping, and the maintenance access ports with port plugs. All of these systems extended the vacuum vessel strong barrier boundary and are potential natural bypasses of the strong barrier. Hence, safety integration became increasingly important with each of the systems that penetrated the strong barrier.

As concepts mature and safety considerations are elucidated during safety integration through design, the areas for safety research and development will be refined. However, there are several areas of safety R&D that are common to all envisioned magnetic fusion energy designs, including tritium retention in irradiated materials, dust generation and entrainment, activation product mobilization, and dispersal of chemically toxic materials. Close correlation to other fusion development themes exist because of the materials characteristics and configurations required for solving the gaps in those themes. The limiting amounts for accidental release of these materials are determined by the adopted safety approach, such as implementation of the no-evacuation criteria. The effectiveness of release mitigation strategies can only be developed, evaluated, and refined through safety integration in design.

MATERIALS LIFE CYCLE MANAGEMENT

Materials life-cycle management and environmental needs are broad and certainly influenced by design configurations, materials selection, and operational performance. Proper handling of the activated materials is important to the future of fusion energy. Fusion offers salient safety advantages relative to other sources of energy, but generates a sizable amount of mildly radioactive materials that tend to rapidly fill the US low-level waste repositories. All three operational commercial repositories will be closed by ~2050 before building the first US fusion power plant. For these reasons and to guarantee the environmental potential of fusion, geological disposal should be avoided. Focus should be placed on more attractive scenarios, such as recycling and reuse of activated materials within the nuclear industry and clearance or free-release to the commercial market if materials contain traces of radioactivity.

There is a need for a global understanding of the activation levels throughout the fusion power core to develop an integrated management approach for all activated materials. Such an approach should consider the radioactivity levels, remote handling issues, property of recycled materials and cost. The key technological needs include the development of 3-D activation codes, remote handling and treatment of tritium-containing materials, separation of materials from complex components, detritiation and tritium capturing and handling, and refabrication of recyclable and clearable materials.

This integral approach calls for major rethinking, education, and research to make the recycling and clearance approaches a reality. In the 2000s, these new approaches have become more technically feasible with the development of advanced radiation-resistant remote handling tools that can recycle highly irradiated materials and with the introduction of the clearance category for slightly radioactive materials by national and international agencies. Note that such approaches are straightforward in application from a technical standpoint, however, a substantial challenge lies in influencing the policy, regulatory, and public acceptance aspects.

The US fusion development program should consider implementing this state-of-the-art recycling and clearance strategy to handle the expected large quantities of fusion-activated materials. A dedicated R&D program would address the issues identified for strategic options. Several areas of scientific advancement underlie sound decisions in restructuring the framework of handling fusion radioactive materials. Examples of specific research needs include development of:

- Low-activation materials that optimize alloying elements specifications to maximize recycling after a short cooling period.

- Techniques and instruments to accurately measure and reduce impurities that deter clearance of in-vessel components.

- Advanced radiation-resistant remote handling equipment capable of handling sizable components with high doses > 10,000 Sv/h.

- 3-D codes that map the activation source term around the torus.

- Efficient detritiation system capable of removing and handling the majority of tritium from activated materials.

- Fusion-specific clearance limits issued by legal authorities; and recycling and clearance infrastructure and market.

Integration of the materials life cycle management strategy into overall fusion development pathways will identify additional research needs.

RELIABILITY, AVAILABILITY, MAINTAINABILITY, AND INSPECTABILITY (RAMI)

Introduction

To successfully harness fusion energy, one must have a product that will operate reliably, with acceptable brief downtimes for maintenance. DEMO is the device on which the performance expectations for future fusion reactors will be based, including its reliability and maintenance characteristics. DEMO must demonstrate a high enough availability for power producers to commit to building a commercial fusion plant. It is not reasonable for power producers to expect an ultimate fusion power plant availability of 80% or more if DEMO cannot demonstrate an availability of 50% or greater. The Priorities Panel recognized the importance of achieving high availability on DEMO with the statement that it must:

> *Demonstrate the productive capacity of fusion power and validate economic assumptions about plant operations by rivaling other electrical energy production technologies.*

Achieving 50% availability on DEMO would be a huge accomplishment. The fission area started with numerous nuclear facilities and the early reactors typically had an availability of 60% or less. It took a number of decades and iterations in design for the technologies and operating procedures to mature and allow the achievement of the 90% availability found in the US today.

In the fusion case, the development path runs from ITER to DEMO, and DEMO is more demanding than ITER in a number of regards, notably:

- The availability must increase by 5 to 10 times.

- The heat and neutron fluxes are doubled.

- The neutron fluence will be more than an order of magnitude greater.

In addition, fusion plants will be more complex than fission plants, requiring many complex systems with remote handling: for example, blankets, divertors, plasma heating and current drive systems, and plasma diagnostics. As was the case in fission development, fusion will require a number of complementary facilities to fill in the technology gaps and evaluate component reliability and lifetimes.

DEMO Requirements

There are three elements involved in achieving the required availability:

- Developing a tokamak design that facilitates reliable operation and efficient maintenance and inspectability.
- Qualifying components to meet their reliability requirements.
- Providing an effective remote handling system and facility design.

Designing for Availability

The main design challenges for availability are the problems caused by 14 MeV neutrons, which damage the materials that surround the plasma and induce radioactivity. After relatively little irradiation, metals can no longer be welded because of helium buildup, and bolts swell. These effects have a profound influence on design because neutrons stream down gaps and can scatter around corners. A rough rule for protecting sensitive areas — such as cooling pipe welds, bolts, in-vessel diagnostics and control components — at the DEMO neutron fluence is that neutrons would have to scatter through a right angle three or more times to reach the area.

Over the years, numerous approaches to designing for high maintainability and availability have been studied: for example, constructing the toroidal system in wedges, and having large ports to allow large components to be removed radially and/or vertically; standardizing components; using redundancy; and having in-service monitoring. Nevertheless, many designs begin to resemble interlocking block, wooden puzzles.

High-level design choices must take into account reliability and maintainability impacts. For instance, liquid metal divertors may be needed to survive plasma transients and provide adequate lifetime. Advanced fuel cycles, e.g., a catalyzed D-D fuel cycle (with tritium sequestration), which avoid the need for complex tritium breeding blankets and have reduced neutron yield promoting long component lifetimes, may be required for power reactor applications. Plasma elongation may be limited by robust plasma position control with a simple, survivable complement of diagnostics. External transform may be required for robust disruption and ELM avoidance.

It is essential:

- To analyze and test approaches for segmented neutron capture and shielding, and remote handling.
- To develop improved integrated designs that satisfy availability and safety requirements by applying lessons learned from ITER, other experiments and remotely handled facilities; and making design choices that promote reliability and maintainability.

Qualifying DEMO Components

Possible failure modes, their failure rate (Fr), and the mean time to repair or replace (MTTR) must be established for every component. In simple terms, the unavailability due to failure is the sum of individual Fr x MTTR. (There is also an unavailability due to components wearing out.) Note that an availability of 50% means there is an unavailability of 50% to allocate to the various power plant components. For example, scheduled maintenance of nonreactor systems in fission typically uses up 5% of the unavailability. This corresponds to allocating 2.5 weeks per year for this activity. The remaining 45% unavailability can be assigned to scheduled maintenance of fusion components and unscheduled downtime.

Maintenance or replacement of fusion's in-vessel components will generally require cooling down the system and opening the vacuum vessel. The time to accomplish this and return to an evacuated state may take weeks, not including the time to repair or replace components. Consequently, the hundreds of in-vessel components (yet to be designed and developed) must be extremely reliable, with long lifetimes. Many of these components will require scheduled replacement after they have accumulated their designed neutron damage. The question is: how will their lifetime and reliability be established on a design yet to be finalized? The complexity of a fusion plant is much greater than that of a fission plant, which makes achieving high availability more difficult. Even when ITER has operated successfully, there will remain gaps in the knowledge and technology base needed for DEMO. Specifically:

- Little reliability or lifetime data that could be extrapolated to DEMO-relevant components — notably for the in-vessel components (the ITER superconducting coils system is much more relevant).
- ITER will provide a valuable test of remote handling techniques, but DEMO needs are more demanding due to the higher degree of complexity and high availability needs.

Consequently, an extensive component testing program, under DEMO-like conditions, will be required to establish the data base of failure rates and maintenance procedures to qualify components for the DEMO. Numerous nuclear and nonnuclear facilities will be required for this task; notably, for plasma facing components, blankets, divertors, and the front ends of heating and current drive systems and diagnostics. As discussed in other parts of this Report, a D-T burning Component Test Facility (CTF) will have to play a key role.

We must assume that during the testing phase some failures will occur, leading to an evolution in component design and in the integrated design itself. This process with a series of incremental improvements is the time-honored approach in developing airplanes and fission reactors.

In the near-term we should undertake a number of activities:

- Determine what we might learn from ITER and other experiments and remotely handled facilities.
- Perform studies to identify the extent of testing beyond ITER needed to qualify components for the integrated design, and ultimately be able to allocate unavailability and evaluate the design for safety.

- Investigate the possibility of performing maintenance on hot components to speed up replacement.

Availability Growth through Remote Handling System and Facility Design

Many view the fusion nuclear environment as the most challenging of the remote handling applications. It is characterized by extreme radiation levels, space-constrained in-vessel access openings, complex and heavy in-vessel components with complex mounting, and service connections that require precision positioning and intricate handling kinematics by robotic mechanisms well beyond today's state-of-the-art technology. The limited in-vessel access typified by tokamak fusion designs is in direct conflict with simple, expedient maintainability. Moreover, fusion reactor concepts include robotic handling and transport of large activated components through the plant facilities, and refurbishment in a hot cell. These operations are unprecedented in themselves — fortunately, ITER is facing them first.

The DEMO integrated design activity must address the remote handling systems and hot cell facilities. Component test facilities, which are proposed to test reactor prototypical in-vessel components, also provide a needed opportunity to develop and test remotely automated and perhaps autonomous maintainable in-vessel components using reactor remote handling systems and hot cell facilities. Preparatory remote handling R&D activities should include:

- Development of large scale, radiation-hard robotic devices that can provide dexterous manipulation and precise positioning of large, multi-ton, highly activated in-vessel components, preferably with simple linear and time efficient motions.

- Development of specialty remote tooling and end-effectors, including precision remote metrology systems to measure plasma facing component alignment, swelling, and erosion in the extreme fusion environment (dusty, high radiation, high temperature, and high vacuum).

- Development of hot cell remote handling systems and tooling necessary to refurbish and/or waste process the activated in-vessel components.

CONCLUSIONS

Designing DEMO to meet the goal of 50% or more availability requires a number of cross cutting development activities:

- Developing and qualifying components in test facilities and monitoring in-service performance in present-day and future machines to develop needed reliability and lifetime data. These test facilities and future CTF-class machines are also required to meet technology development needs beyond our area.

- Initiating an aggressive reliability growth and maintainability improvement program that builds on what we learn from testing and operating components.

- Initiating an integrated design activity for DEMO that will process what we learn from ITER, test facilities, and future CTF-class machines and develop a credible, low risk, attractive design for DEMO.

- Developing the design of an efficient remote handling system and hot cell facility as part of the integrated design activity, performing preparatory remote handling R&D, and prototyping on CTF-class machines.

Thrusts

The many research requirements identified for this Theme will be addressed by the set of Research Thrusts. In many cases, the research requirements are addressed in multiple thrusts. These are summarized below, with the thrusts linked to each panel listed in approximate order of their connection to the panel. Details of each thrust are given in Part II of this Report.

PANEL	RESEARCH THRUST	COMMENTS
Fusion Fuel Cycle	**Thrust 13:** Establish science and technology for fusion power extraction.	Primary thrust to develop model and progressively more integrated experiments to determine if fuel sustainability can be achieved (including tritium breeding and tritium fuel cycle processing in a practical system).
	Thrust 14: Develop science and technology needed to harness fusion power.	Development of functional materials including for tritium breeding.
	Thrust 15: Create integrated designs and models for fusion power systems.	Design integration and modeling of all major functions and processes for a power plant.
Power Extraction	**Thrusts 2 and 4:** Control transient events in burning plasmas; Qualify operational scenarios and supporting physics basis for ITER.	Determine the plasma conditions associated with transient events and plasma control that will guide the requirements for the design and function of power extraction components during normal and transient plasma operations.
	Thrust 11: Improve power handling through engineering innovation.	Support power extraction techniques and predictive capabilities specifically tailored for first wall and divertor surface heat loading compatible with blanket integration requirements.
	Thrust 13: Establish science and technology for fusion power extraction.	Primary thrust to develop model and progressively more integrated experiments to achieve efficient power extraction with concurrent tritium self-sufficiency.
	Thrust 14: Develop science and technology needed to harness fusion power.	Development of structural and tritium breeding materials in support of power extraction component/subsystem requirements.
	Thrust 15: Create integrated designs and models for fusion power systems.	Design integration and modeling of all major systems and processes for a power plant (with power extraction as a major aspect).

PANEL	RESEARCH THRUST	COMMENTS
Materials Science	**Thrust 1:** Develop measurement techniques to understand burning plasmas.	Support development of the means to control burning plasmas by developing fundamental radiation effects knowledge on plasma diagnostic materials.
	Thrust 7: Exploit high temperature superconductors and magnet innovations.	Support development of advanced magnets by providing fundamental radiation effects knowledge on high-temperature superconducting materials.
	Thrusts 9, 10, 11, 12: Thrusts for plasma materials interactions theme.	Support development of plasma facing materials and structures resistant to plasma interactions and radiation damage.
	Thrust 13: Establish science and technology for fusion power extraction.	Develop structural and tritium breeding materials in support of power extraction component and subsystem requirements.
	Thrust 14: Develop science and technology needed to harness fusion power.	Primary thrust to develop basic materials properties information and models on the behavior of materials in the harsh fusion environment.
	Thrust 15: Create integrated designs and models for fusion power systems.	Support integrated design and modeling of fusion power systems by linking fundamental materials models, structural models and design codes.
Safety and Environment	**Thrust 13:** Establish science and technology for fusion power extraction.	Understanding tritium behavior in power extraction concepts is crucial for ensuring safe operation of the fusion reactor and fuel handling systems.
	Thrust 14: Develop science and technology needed to harness fusion power.	Activation from neutrons and retention of tritium in materials influence safety and environmental considerations. Developments in fusion materials sciences can inform aspects of materials behavior in off-normal and postulated accident conditions, as well as long-term materials life-cycle management.
	Thrust 15: Create integrated designs and models for fusion power systems.	Primary thrust for assessing and conducting studies of major safety and environment aspects within an integrated design activity.
Reliability, Availability, Maintainability, and Inspectability (RAMI)	**Thrust 13:** Establish science and technology for fusion power extraction.	Primary thrust for implementing an aggressive program for reliability growth.
	Thrust 15: Create integrated designs and models for fusion power systems.	Primary thrust for assessing and conducting studies of RAMI aspects within an integrated design activity.

HARNESSING FUSION POWER
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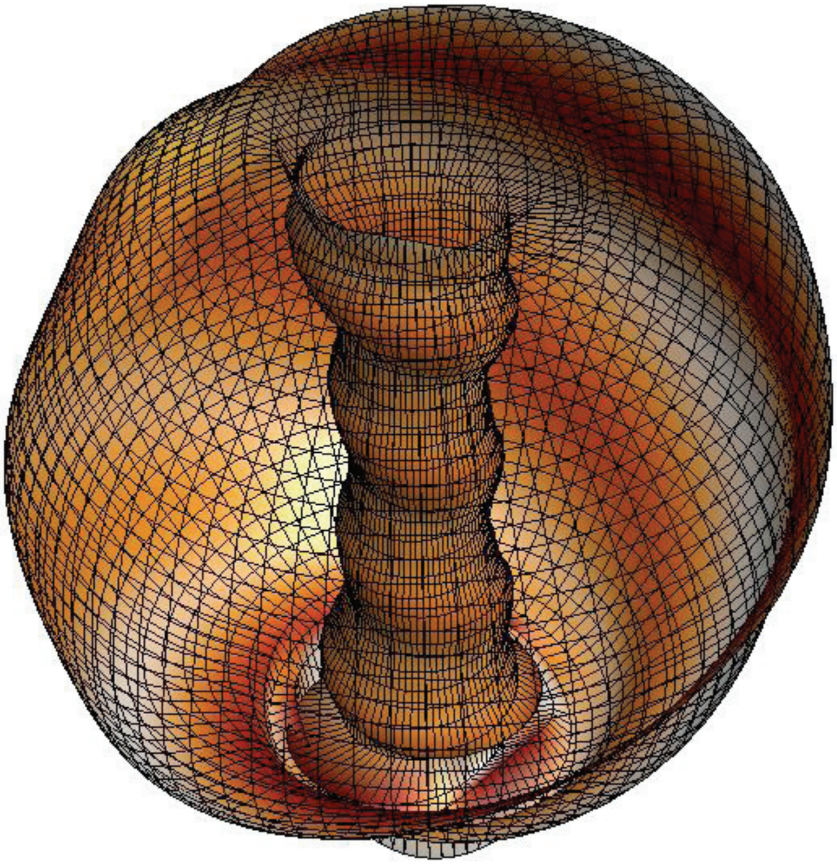
THEME 5:

OPTIMIZING THE MAGNETIC CONFIGURATION



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Reconstruction of plasma distortion associated with a resistive wall mode in a spherical torus.
(From S.A. Sabbagh, Columbia U.)

THEME 5: OPTIMIZING THE MAGNETIC CONFIGURATION

Introduction

SCOPE AND FOCUS

A key strategy for the US and world fusion program has been the investigation and development of a variety of magnetic configurations that are able to confine a hot, stable plasma. Of these, the tokamak configuration has achieved parameters that are nearest those required in a fusion reactor. The tokamak exhibits good qualities for plasma confinement, and this has spurred its intense development, ultimately leading to ITER. Nevertheless, the optimum magnetic configuration for solving the simultaneous challenges of fusion plasma physics and engineering is not yet known. Each of the magnetic configurations being studied brings opportunities to optimize the magnetic fusion system in unique ways. Each also has challenges that reflect tradeoffs in physics and engineering.

As part of its strategic planning process, the Department of Energy asked the Fusion Energy Sciences Advisory Committee (FESAC) to critically evaluate the status of, and scientific opportunities for, major alternate magnetic confinement configurations. This led to the Report of the FESAC Toroidal Alternates Panel (TAP), which forms the starting point for the Optimizing the Magnetic Configuration Theme covered in this chapter. (The TAP report is available at http://fusion.gat.com/tap/). In this context, "alternate" means toroidal magnetic configurations other than the conventional aspect ratio tokamak. The scope of the evaluation was limited to the toroidal magnetic configurations that are most developed: the **stellarator**, the **spherical torus**, the **reversed field pinch**, and **compact tori**. Compact tori are two distinct configurations, the spheromak and the field-reversed configuration, that share the common feature of having a simply connected wall geometry with no physical structure linking the plasma torus.

The toroidal alternates offer particular scientific and technological benefits that could substantially improve the vision of a fusion reactor, if their physics challenges can be met. Indeed, this is a primary motivation for research on alternate configurations. While the nomenclature used to identify specific toroidal configurations is important for historical and programmatic reasons, the tokamak and toroidal alternates are in reality closely related. Together they span an essentially continuous spectrum of magnetic configurations, varying in major variables such as the aspect ratio of the torus, degree of internal versus external magnetization, and degree of 3-D shaping of the magnetic field. Figure 1 illustrates this configuration space. The axes are the major variables noted above, and "spots" are located at the combination of variables that represent the typical or historical view of the named configurations. This is not intended to be an exact representation of the configurations, rather a visual to help expose the opportunity and need to understand and predict toroidal confinement over a wide range of parameters. Continuing exploration of this configuration space is a central idea of ReNeW Research Thrusts 16-18.

Figure 1 also illustrates that multi-configuration fusion research advances the understanding of fusion plasma physics much more completely than can be achieved by any one configuration on its own, by broadening the scientific approach to grow and validate fusion science over a wide range

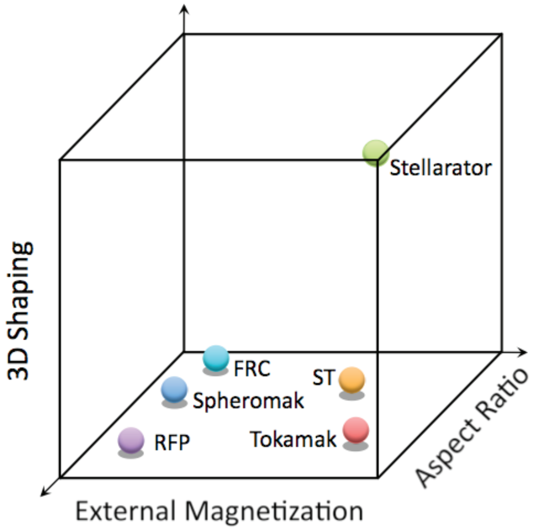

Figure 1. Configuration space for toroidal magnetic confinement.

of plasma conditions, and thereby enhancing the opportunity for scientific discovery and innovation in toroidal confinement. It is possible that the optimum magnetic configuration will have features and qualities that are significantly different from any of those studied in present-day devices, including the tokamak.

For each of the four alternates, the TAP identified a research goal for the ITER era, roughly the next 20 years. The TAP also identified the critical research issues and gaps in knowledge for each of the magnetic configurations. The organization of the Optimizing the Magnetic Configuration Theme inherited the structure of the TAP process by creating a panel for each of the alternate configurations. The issues and gaps identified by the panel formed the basis for elaborating the research needs that appear later in this Chapter.

HIGHLIGHTS OF ACCOMPLISHMENTS IN TOROIDAL ALTERNATE CONFINEMENT RESEARCH

Significant progress has been made in toroidal alternate research in recent years. Each configuration is described briefly, and highlights of research accomplishments noted.

Stellarator

The stellarator uses nonplanar magnets to produce nearly all of the helical magnetic field required for toroidal confinement, thus reducing or eliminating the need for a plasma current. Stellarators are intrinsically steady state, and they do not exhibit current-driven instabilities that lead to fast disruptive termination. The magnetic field has a visibly large degree of 3-D shaping, but the field can be designed with “quasi” symmetry to attain similar confinement properties of axisymmetric plasmas. Highlights of stellarator research include:

- Many stellarator devices have been successfully built and operated. Fusion plasma parameters achieved are second only to tokamaks. Ion temperatures of 7 keV have been attained, and energy confinement is similar to tokamak scaling. Advanced quiescent confinement modes have been obtained without impurity ion accumulation.

- Stellarators designed with quasi-symmetric magnetic fields have demonstrated reduced neoclassical transport in low collisionality plasma regimes.
- A peak plasma pressure of 1.5 atmospheres and (separately) a volume-averaged normalized plasma pressure of $\langle\beta\rangle$ >5% have been achieved. (β is the ratio of plasma pressure to magnetic pressure.) In contrast to tokamaks, the limiting behavior for plasma pressure is benign and does not lead to an abnormal termination of the plasma.
- Maximum plasma densities achieved in stellarators are well above those in tokamaks at comparable magnetic field strengths. The density limit is understood in terms of a simple radiative power balance. Exceeding the density limit does not result in abrupt plasma termination, but a relatively slow radiative collapse.
- Near steady-state operation has been attained, with record energy throughput for any toroidal plasma.
- Application of 3-D magnetic fields has been shown to control edge localized modes (ELMs) in tokamaks, and to induce plasma rotation through neoclassical transport effects.

Spherical Torus

The spherical torus (ST) is a low aspect ratio tokamak, i.e., the ratio of the plasma's major radius to minor radius is relatively small (< 2). A principal advantage of the ST is stability provided by enhanced magnetic field line curvature. This stability at high plasma pressure, along with the more spherical-like geometry, could lead to a compact fusion system with simple magnet design and good maintainability. For these reasons, the ST is considered a strong candidate as the basis for a nuclear fusion component testing device. Highlights of ST research include:

- Noninductive plasma initiation and growth: Startup plasma currents have reached 25% of the required level (160kA using coaxial helicity injection [CHI], and 100kA with point-source helicity injection). Post-formation current buildup (to 350 kA) was shown in steady-state CHI discharges. Robust transition to inductive current drive was demonstrated.
- Plasma-material interface in compact geometry: A significant increase in energy confinement and an ST-record plasma thermal energy (0.48 MJ) were achieved with the application of lithium to plasma facing components. A four-fold reduction of the peak divertor heat flux was shown via gas injection while maintaining high-energy confinement.
- Confinement at low aspect ratio: A strong dependence of energy confinement on the toroidal field strength was found, in contrast to conventional tokamak behavior. An onset of short wavelength turbulence is consistent with drift wave theory for the electron temperature gradient instability. A 40-50% reduction in global recycling and a five-fold increase in electron energy confinement were demonstrated in a small device with liquid lithium walls.
- Stability of high beta ST plasmas with very broad current profiles: Volume-averaged plasma pressure, $\langle\beta\rangle$, up to 20% was produced. (This corresponds to a toroidal beta value

of 40%.) Localized instabilities (tearing modes, edge localized modes) were mitigated or eliminated via lithium application, and record stability levels (normalized beta of 7.2) were reached. Kinetic theory has provided understanding of the complex relationship between global instabilities and plasma rotation. The connection between observed fast particle loss and toroidal Alfvén instabilities/theory was established.

- Control of high beta ST plasmas: Active control of resistive wall modes and maintenance of favorable plasma rotation profiles decreased the disruption probability for long-pulse plasmas while exceeding the stability limit for nonrotating plasmas by over 50%. Rotation is controlled by 3-D magnetic fields in plasmas with strong momentum input.
- Integration and sustainment at low aspect ratio: Stability levels approaching that needed for a fusion component testing device were sustained for significant duration (3-4 resistive diffusion times) with a majority of the plasma current provided noninductively.

Reversed Field Pinch

The reversed field pinch (RFP) is an axisymmetric configuration in which electrical current in the plasma generates most of the magnetic field. The requirements on the external magnets are therefore greatly reduced. The RFP has a high ratio of plasma pressure to magnetic pressure. The ohmic heating from the plasma current could reduce or eliminate the need for complicated auxiliary heating systems. These features could yield a high power density fusion energy system with simple construction. Highlights of RFP research include:

- A 10-fold increase in energy confinement is obtained when fluctuations in the magnetic field are reduced using active control of the plasma current profile. This confinement is comparable to that in a tokamak of the same size and plasma current. High-temperature plasmas are produced without auxiliary (non-ohmic) heating, with electron temperature T_e ~ 2 keV and ion temperature T_i ~ 1 keV.
- High-current plasmas (up to 1.5 MA in RFX-mod) spontaneously generate a helical magnetic field in the plasma core. The energy confinement time is improved in these self-organized, quasi-helical plasmas. The confinement time is increasing with plasma current.
- Plasmas with thermal pressure exceeding theoretical limits for interchange and magnetic tearing stability are produced without exhibiting fast disruptive behavior.
- Active feedback stabilization is now routine when the plasma is contained in a thin-walled metal shell. Discharge lengths up to 0.5 s are obtained, which is more than ten growth times for thin-shell instabilities that occur without feedback control.
- Up to 10% of the plasma current has been sustained using low frequency AC induction (helicity injection), a promising approach for efficient steady-state operation of the RFP.
- The physics of magnetic self-organization has been advanced by RFP (and compact torus) research, with strong connections to plasma astrophysics. Areas include dynamo processes, momentum transport, collisionless ion heating, and magnetic reconnection and turbulence.

Compact Torus

A compact torus (CT) is a toroidal plasma configuration, but no physical structure threads or links the hole in the plasma. Compact tori are formed in cylindrical chambers, greatly simplifying assembly and maintenance. There are two distinct CT types: the (i) field-reversed configuration (FRC) and the (ii) spheromak. The FRC has the highest possible value of beta (ratio of thermal plasma pressure to magnetic pressure) for confined plasmas. It is diamagnetic, with a plasma current perpendicular to the magnetic field. The spheromak has moderately high beta, and like the RFP, the plasma current generates most of the magnetic field. These features could lead to compact fusion systems with properties that promote high availability. Highlights in CT research include:

FRC

- High-density, high-temperature FRCs have been routinely produced in field-reversed theta-pinch devices. In the Large *S* experiment, a 1.0×10^{21} m^{-3} density, 0.575 keV electron temperature, and 1.225 keV ion temperature were simultaneously obtained in a transient discharge; in the FRX-L experiment, much larger densities at lower temperature were obtained.

- FRCs that overcome impurity radiation barriers and have $T_e+T_i>$ 0.2 keV have been produced and sustained using low-megawatt, steady-state power. The required excellent vacuum environment has been obtained via baking and glow discharge cleaning. Titanium gettering has eliminated uncontrolled recycling so that FRCs can now be fueled by gas puffing. (TCSU experiment).

- FRCs formed and sustained by rotating magnetic fields are found to have central conductivity ~5 times higher than the average, edge-dominated value. (TCSU experiment).

- Electron temperatures > 0.3 keV have been sustained for >3 ms in small plasma columns heated by <10 kW of rotating magnetic field (RMF) power. It was discovered that odd-parity RMF can both keep field lines closed, and heat ions and electrons to high temperatures. A fully electromagnetic, 3-D Particle-in-Cell code has modeled FRC formation. (PFRC code).

- Successful formation and inductive sustainment of oblate FRCs have been achieved due to complete stabilization of the tilt mode and improved stability to other magnetohydrodynamics (MHD) modes by external field shaping and a central conductor. (MRX experiment).

- Numerical simulations have demonstrated oblate FRC stabilization using a combination of close-fitting conducting shell and neutral beam injection effects. (HYM code)

Spheromak

- Significant progress has been achieved in performance: By using helicity injection for both formation and steady-state sustainment, spheromaks with 1 T toroidal magnetic field, 1 MA toroidal plasma current and $1x10^{20}$ m^{-3} density have been produced. Transient electron temperatures as high as 0.5 keV were obtained in the low-turbulence, decaying phase of spheromak plasmas. (CTX, SSPX experiments).

- Reduction of spheromak magnetic turbulence has improved confinement. Best confinement has been obtained by tailoring the current profile to avoid low-order rational surfaces. This was accomplished by edge direct current injection at a level slightly below the sustainment value. The result was core electron energy confinement consistent with tokamak L-mode scaling. (SSPX experiment).
- Whole-device, resistive MHD simulations (NIMROD code) agreed well with experiments (SSPX) and improved understanding of the coupling between transport and nonlinear 3-D magnetic evolution.
- Spheromaks have been formed and sustained by the continuous injection of magnetic helicity from an inductive source. This theoretically predicted method has achieved a 34 kA toroidal plasma current, which is nearly double the injected current. (HIT-SI)
- New experiments have begun to explore innovative startup and current-drive techniques. (PBX, LANL-DRX experiments).
- Basic physics experiments have advanced the understanding of the high-speed dynamics intrinsic to helicity injection (Caltech) and the understanding of magnetic reconnection and relaxation processes (SSX, RSX experiments).

Research Requirements

The research requirements associated with each of the toroidal alternates are described. A more detailed discussion of the scientific issues is included in the FESAC Toroidal Alternates Panel Report.

THE STELLARATOR

Introduction

The stellarator is a toroidal device that nominally produces all of the confining magnetic field through use of external coils. An immediate advantage of the configuration is intrinsically steady-state operation because there is no need for a pulsed transformer to drive a plasma current. Stellarators, having no need for noninductive current drive, could prove to be very efficient fusion systems. Plasma startup is on existing magnetic surfaces and positional control is not needed. Sophisticated plasma control to avoid discharge-terminating current-driven instabilities is largely eliminated.

Production of the confining fields with external currents requires that the stellarator be an inherently 3-D configuration. The 3-D shaping of the plasma provides for a broader range in design flexibility than is achievable in a 2-D system. Also, in a steady-state 3-D system there is no need to design for large transient thermal and mechanical loads on the vacuum vessel and supporting structure associated with abrupt confinement termination (major disruptions). However, as shown for example in Figure 2, coil systems are more highly shaped than simple toroidal and poloidal field coils.

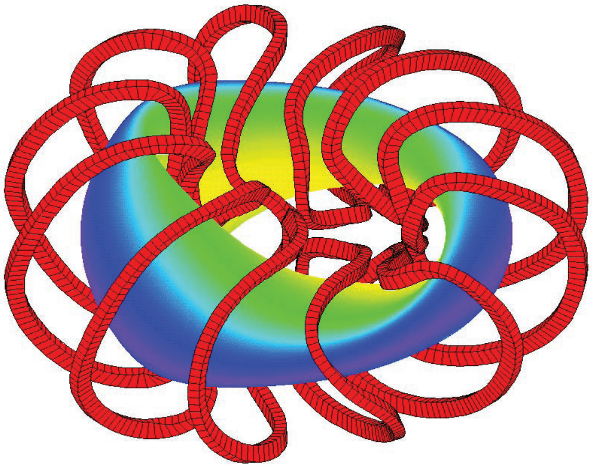

Figure 2. Outer surface of a stellarator plasma surrounded by the nonplanar magnet coils that produce the plasma configuration within (courtesy of P. Garabedian[1]).

Although more complex, many stellarator systems have been successfully built and operated. Plasma parameters achieved to date are second only to tokamaks. Large stellarators are pursued overseas: LHD in Japan (operating) and W7-X in Germany (2014). The 3-D nature of stellarators can lead to unacceptably large thermal transport in reactor-relevant regimes and poor confinement of the energetic fusion products. As a result, stellarators must, and can, be optimized through 3-D shaping to eliminate these losses. The W7-X design was optimized by aligning particle drift orbits with magnetic surfaces and having minimal plasma currents. The US program is focused on optimization through quasi-symmetry (symmetry in the magnitude of the magnetic field in a particular direction). Quasi-symmetry ensures the same good particle orbits that true symmetry provides in the tokamak. The US is a world leader in this concept, and its viability has been demonstrated in hot electron plasmas in the university-scale Helically Symmetric Experiment (HSX). Quasi-symmetry allows both lower flow damping (important for reducing turbulent transport) and smaller plasma aspect ratio than in W7-X, and thus provides a potentially improved solution for steady-state, stable, thermal plasma containment, with good confinement of energetic fusion products.

Quasi-symmetry in a torus can be achieved in either the toroidal direction (long way around the torus; known as quasi-axisymmetry or QA), the poloidal direction (short way around the torus; known as quasi-poloidal symmetry or QP), or in a helical direction (known as quasi-helical symmetry or QH). All of these quasi-symmetries are predicted to ensure well confined particle orbits. Significant differences among these configurations do, however, exist. As one example, QA has a significant toroidal current driven by the plasma pressure gradient (bootstrap current) in contrast to similar plasmas in QH or QP configurations. Other differences include global and local magnetic shear, field curvature, location and fraction of trapped particles, magnetic well depth, and plasma viscosity. A proper selection of type of quasi-symmetry and magnetic field optimization may provide passive control of not only collisional transport and energetic particle confine-

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[1] *P. Garabedian, Proc. Natl. Acad. Sci. USA 104, 12250 (2007).*

ment, but also micro- and macro-instabilities – that is, turbulent transport and global stability. The different approaches might also vary in optimal aspect ratio, how they simplify the coil structures, or how they approach a 3-D divertor design.

The promise of stellarators is well recognized within the fusion community. Stellarator research is a mature field with many experiments worldwide including large performance extension facilities. Recommendation 4 in the FESAC "Priorities, Gaps, and Opportunities" report (PGO) urges consideration of a major initiative (I-5) for an "Advanced experiment in disruption-free concepts," of which the stellarator would be the most likely facility. The FESAC TAP report states: "There is little doubt that a stellarator configuration can confine plasma at the parameters necessary for fusion burn" and that stellarator research "is likely to contribute significantly to the optimization of tokamaks and perhaps to other confinement concepts." This motivated the stellarator ITER-era goal reported by the TAP:

Develop and validate the scientific understanding necessary to assess the feasibility of a burning plasma experiment based on the quasi-symmetric (QS) stellarator.

A critical gap exists in the worldwide stellarator program in that there are no moderate scale experiments investigating quasi-symmetry. There is an immediate need to explore quasi-symmetry with sufficient breadth and scale to achieve the ITER-era goal.

Scientific Issues and Research Requirements

INTEGRATED HIGH PERFORMANCE OF QUASI-SYMMETRIC OPTIMIZED STELLARATORS

There is a need to study a plasma in a quasi-symmetric stellarator that has both high plasma pressure and low ion collisionality. Such a test of quasi-axisymmetry was to be performed on the National Compact Stellarator Experiment (NCSX), which was under construction until 2008. Its cancellation has left a gap in the worldwide program of exploring the science of quasi-symmetry and the potential of the concept. As described earlier, quasi-symmetric stellarators may differ in how they approach neoclassical and turbulent transport as well as macrostability. Therefore, there is a need for multiple quasi-symmetric configurations to assess high-beta, low collisionality, $T_i \sim T_e$ plasmas with good confinement. Because of the centrality of the issue of disruption control, one approach is to differentiate the experiments based on the amount of bootstrap current present for a given set of plasma parameters. The configurations will differ in other attributes such as divertors or impurity control. Based on the results of the multiple experiments, theory and modeling, information gained from the international stellarator program, a performance extension (PE) level quasi-symmetric stellarator that extrapolates to burning plasma performance would be designed and constructed.

Research Requirements

The multiple quasi-symmetric stellarators would have a common set of research requirements. These requirements serve as definitions of "integrated high performance." Implicit within these requirements is sufficient operational flexibility of the magnetic geometry and plasma parameters to test theory and modeling predictions.

- Sufficient power to test the configuration's operational limits with respect to beta with the constraint that the ions are simultaneously in the low collisionality limit.
- Sufficient pulse length for the magnets and heating supplies so that the bootstrap current can reach at least near-equilibrium to test susceptibility to disruptions.
- A divertor that is compatible with quasi-symmetry and that can prevent radiative collapse of the plasma at high density and test the limit of high-density performance.
- Capability to explore the confinement of fast ions to understand whether it is possible to reduce energetic fusion product losses in a quasi-symmetric fusion reactor.
- Capability to study impurity generation and transport to determine whether low-impurity regimes are accessible with quasi-symmetry and are reactor relevant, especially without edge localized modes.
- Capability to investigate the role of plasma flow and its effect on anomalous transport. Other approaches to minimize anomalous transport might be gained from gyrokinetic studies to find configurations in which the heat flux is only weakly dependent on the temperature scale length.
- Capability to obtain data on energetic particle instabilities from radiofrequency or neutral beam-driven fast ions.

DESIGNING COILS FOR 3-D SYSTEMS

The basic principle of stellarator design is that physics targets determine the parameters of the field and resulting coil structures. Relaxation of the physics constraints can permit improvements in the coil engineering. For example, experiments, especially in low current configurations, suggest that stability to ballooning modes may be unnecessarily restrictive as a design target.

The existence of error fields is unavoidable; they can be introduced through random or systematic winding variations or placement and assembly errors of the coils into a torus. Trim coils are routinely used in both tokamaks and stellarators to trim out errors. There is a need to explore increased use of trim coils to reduce the large cost impact of overly accurate fabrication and assembly. The effects of this trimming on the desired field structure and plasma performance, including thermal and alpha confinement and flow damping, needs to be considered.

Other factors to be considered are aspect ratio and divertors. Coils become less distorted at higher aspect ratio, while divertors may drive additional complications. Metrics need to be identified to quantify improvements and satisfaction of design goals. Lessons learned from recent construction experience with NCSX and W7-X need to be considered.

Three-dimensional fields may be applied to axisymmetric systems in different forms. Small non-symmetric fields (resonant magnetic perturbations — RMPs) have been applied to tokamaks for ELM suppression. Quasi-axisymmetric shaping could be used for control of disruptions and vertical instability. Three-dimensional shaping might be applied to the reversed field pinch to reduce the power threshold for achieving the good confinement quasi-single helicity state (QSH).

Research Requirements

- Evaluate the physics targets for device optimization and the sensitivity of the designs to these targets. Identify the level of quasi-symmetry needed as reactor conditions are approached.
- Identify the most deleterious field errors and how to correct with trim coils.
- Determine optimal aspect ratios and divertor solutions for QS stellarators; this may depend on the type of quasi-symmetry employed.
- Examine the use of magnetic materials for field shaping and coil simplification.
- Investigate how 3-D fields can be used to improve performance in axisymmetric devices.

PREDICTIVE CAPABILITY

Improved predictive capability is a key element across all of toroidal confinement and is the basis for Thrust 6. Three-dimensional equilibrium effects are central to stellarator optimization. There is a need to further develop major theory and computational efforts that have the ability to describe plasma physics in 3-D configurations without well-formed magnetic surfaces. Equilibrium determination and finite-beta magnetic field structure are overlapping issues for all toroidal configurations. This is especially true given that stability limits in stellarators appear to manifest themselves as degradation in transport and not virulent behavior. It would be highly desirable for major simulation codes in the US fusion program to have the capability to handle 3-D configurations.

Research Requirements

- Predictive tools that credibly describe 3-D equilibria including the effects of magnetic islands, plasma flow, stochasticity, and energetic particle modes.
- Clear understanding with experimental confirmation of stability limits in stellarators.
- Three-dimensional equilibrium reconstruction capability.
- Integration of 3-D effects into the Fusion Simulation Program (FSP) in the US.

DIVERTORS

A key element for any fusion device is an effective divertor to remove heat and helium "ash," and to restrict impurity atoms from entering the core plasma. Divertors have been essential in achieving present stellarator results, and led to record hour-long discharges in LHD, limited only by divertor power handling. A radiative divertor with a more uniform power deposition is plausible in advanced stellarators because they can operate with high-edge densities. Nonetheless, divertor development for stellarators trails that for tokamaks, and is furthermore complicated by the 3-D geometry. Much of the work on stellarator divertors makes use of the island divertor concept, which depends on edge resonances in the rotational transform. This introduces a need to control the edge magnetic geometry, and the interaction region is undesirably close to the confinement region.

Research Requirements

The development and understanding of effective 3-D divertors for stellarators is a high-priority research need. Essential research requirements are:

- Modeling to determine how the interaction region can be expanded and/or removed from proximity to the fusion plasma and reduce power levels and impurity generation to acceptable values.
- Experimental validation of divertor and edge modeling codes.
- Design of divertor structures integrated into stellarator design codes to preserve the quasi-symmetry of the configuration (at least in the core).
- Collaboration on divertor experiments on LHD and W7-X.

IMPURITY AND FUSION ASH ACCUMULATION

Impurity accumulation has been observed experimentally in stellarators in some regimes, especially at high density with improved confinement. To avoid radiation collapse for high-density, long-pulse operation, it is important to screen the impurity influx at the edge and degrade the impurity confinement in the core. Without a large pedestal, ELMs may not be available to prevent impurity accumulation in a stellarator. However, very high-density, low-impurity stable discharges without ELMs have been achieved during Super Dense Core (SDC) plasmas in LHD and high-density H-mode (HDH-mode) plasmas in W7-AS. Neoclassical calculations indicate that for a tokamak in the banana collisionality regime the ion temperature gradient promotes impurity expulsion. In contrast, for a conventional stellarator at low collisionality, impurities are predicted to accumulate in the core. It is presently unknown how much symmetry-breaking is allowable before temperature screening of impurities is defeated. There is presently no experiment that can test impurity transport physics for quasi-symmetric configurations at low ion collisionality. An experimental test is needed to determine whether temperature-screening or anomalous transport can sweep the impurities out of the plasma.

Research Requirements

- Understanding of impurity transport in quasi-symmetric configurations with low enough charge exchange losses to support an ion temperature gradient. Such experiments would study the role of ELMs and magnetic islands in impurity accumulation and the influence of a divertor on impurity transport. The possibility of a radiative divertor should also be explored.
- Gyrokinetic and neoclassical calculations of impurity and helium ash transport for stellarators, especially to search for configurations that expel impurities.
- Participation in collaborative experiments on impurity transport in LHD and W7-X can aid understanding of techniques to avoid impurity accumulation.
- To study temperature screening in a stellarator or PE class experiment, comparable to DIII-D, may be required. The required plasma conditions are: bulk and impurity ions in

- the banana regime, low turbulent transport, small density gradient, large ion temperature gradient, and long pulse time to adequately capture impurity transport.
- Exploration of helium ash removal on an intermediate or PE class experiment. However, D-T operation may ultimately be needed to adequately resolve this issue.

OPERATIONAL LIMITS

It appears that typically neither density nor beta is limited by either MHD instabilities or disruptions in low-current stellarators. In both LHD and W7-AS, the achievable β was limited by the available heating power, with no hard β-limit seen. MHD activity seen in the experiments is consistent with the theoretical predictions, but the modes saturate and they do not impede access to higher $\langle\beta\rangle$ values. Three-dimensional equilibrium codes that do not assume the existence of nested flux surfaces find a substantial region of stochastic (wandering) magnetic field lines in the outer region of the plasma at the highest values of β in both W7-AS and LHD.

Densities of 4×10^{20} to 10^{21} m^{-3} have been achieved in W7-AS and LHD. Without large plasma currents there is minimal magnetic energy dissipated at termination. Abnormal termination of stellarator discharges does take place, however, when densities become too large. The maximum achievable plasma density in a stellarator is determined by a soft radiative collapse on the confinement time scale when there is balance between the heating power and radiation losses.

Research Requirements

- Develop a theoretical understanding of the beta limit in stellarators.
- Understand at what level of plasma current are there tokamak-like limits with respect to pressure and density.
- Determine discharge termination behavior with significantly more poloidal field due to large plasma currents in QS configurations.
- Model and validate the deterioration of magnetic surfaces with plasma pressure; this may be a crosscutting issue for all of toroidal confinement.

ANOMALOUS TRANSPORT REDUCTION

Quasi-symmetric stellarators have reduced neoclassical transport and consequently turbulent-driven transport becomes dominant. This has been demonstrated experimentally in HSX. Furthermore, a number of theoretical and experimental results suggest that the reduction of neoclassical transport will also result in the reduction of turbulent transport. Quasi-symmetric stellarators have a minimum in damping due to parallel viscosity in the direction of symmetry. This opens the possibility that strong flow shear or reduced zonal flow damping may reduce turbulent transport in stellarators. HSX can examine flows and anomalous transport in a quasi-symmetric stellarator with a hot electron plasma. Turbulence and transport in hot ion plasmas can presently only be studied in nonsymmetric and quasi-omnigenous stellarators such as LHD and W7-X. Close collaboration between theory and experiment can help to understand why stellarator profiles, under some conditions, may be less stiff than for tokamaks.

Research Requirements

- New quasi-symmetric experiments are needed to study turbulence and transport barrier formation in a hot ion, high-beta plasma. Quasi-symmetric stellarators optimized to reduce both neoclassical and turbulent transport are needed.
- Collaboration on turbulence and modeling with LHD and W7-X.
- Nonlinear gyrokinetic calculations are needed to understand zonal flow formation and critical gradients in stellarators. The role of magnetic shear and E x B stabilization needs to be addressed specifically for stellarators as has already been done for tokamaks.
- Neoclassical calculations are needed to understand whether transport barriers are possible in quasi-symmetric stellarators due to electron/ion root proximity.
- The application of 3-D shaping to tokamaks with respect to intrinsic rotation to control turbulence and resistive wall modes would be useful. This might be done most effectively through the modification of an existing tokamak.

ENERGETIC PARTICLE INSTABILITIES

Alfvénic instabilities and energetic particle modes (EPMs) are frequently observed in high-performance tokamak regimes, and have also been observed in stellarators. Since stellarators encompass a wide range of magnetic geometries and symmetries, a larger set of Alfvén instabilities can be present than in tokamaks. Fast ion driven instabilities have been observed for all stellarator experiments in which energetic tail heating populations are present. Regimes have been observed both where the effects of such modes are relatively benign, as well as cases where they lead to lowered core confinement. Stellarators offer an effective means for suppressing these instabilities through high-density operation. It is important to understand how lowered core confinement may prevent the path to high-density operation.

Research Requirements

- Continued refinement in spatial, velocity and time resolution of fast ion diagnostics will be essential in understanding these phenomena and their implications and control in future devices.
- Further improvements will be needed in the theory and modeling of fast ion instabilities in stellarators in mode identification, linear thresholds, and fast ion loss prediction, along with a greater focus on the nonlinear physics, to attain the desired predictive capabilities. Experimental validation is essential.
- Since the form of fast ion instabilities can be influenced by the 3-D magnetic field structure, a target for energetic particle instability suppression should eventually be included in stellarator physics optimization efforts.

DISRUPTIONS AND PROFILE CONTROL

Disruptions in low-current stellarators do not occur in typical operation, even at plasma pressures above the calculated ideal stability limits. Disruptions can be forced by fast current-ramps in stellarator discharges with ohmically-driven current. Disruption avoidance in stellarators is thus generally not a critical challenge. It is not yet clear, however, what level and distribution of current can be tolerated in a high-β stellarator while remaining robustly stable.

It is possible that the application of 3-D quasi-axisymmetric shaping to tokamak-like plasmas may confer some disruption immunity and reduce the requirements for rapid control. It is not known what level and type of 3-D shaping might suppress disruptions in tokamak plasmas without adverse effect on confinement. Experimentally, information could potentially be obtained through 3-D shaping experiments/modifications to smaller tokamaks, which could hopefully be flexible to provide a wide range of data.

Stellarators do not have stringent control requirements for maintaining the density and temperature profiles for optimal bootstrap currents for stable, sustained operation. Thus, the allowable plasma profiles in stellarators are flexible and may be used to advantage, e.g., high-density edge plasmas for radiative divertor operation or smaller edge gradients for ELM suppression. Peaked density profiles due to reduced thermodiffusion (associated with the good confinement) may, however, result in impurity accumulation.

There are presently no experiments with strong 3-D shaping also capable of large ohmic or bootstrap currents. A university-scale stellarator experiment is studying disruption suppression with stellarator fields, but has only a modest ohmic heating capability. The extent to which non-axisymmetric 3-D fields can be applied to suppress disruptions, as well as for general concept improvement in tokamaks, remains to be fully examined and is an important area for investigation. Because stellarators already demonstrate the passive avoidance of disruptions, it is clearly a requirement to maintain this property through validated understanding. Any high-performance stellarator designs, including QA stellarators with comparatively high levels of bootstrap current, must be evaluated for susceptibility to disruption and the related need for profile control.

Research Requirements

- Ensuring that higher performance quasi-symmetric stellarator plasmas with significant bootstrap current continue to passively avoid disruptions or, if not, determining the disruptive boundaries.
- Understanding the transition of stellarators away from disruption immunity with application of significant current.
- Extension of variable levels of 3-D shaping to tokamaks to avoid density and pressure-driven disruptions.
- The benefit of 3-D shaping to high-performance tokamaks would most likely require an experimental test on a high-performance tokamak once the appropriate factors have been identified from smaller experiments, theory and modeling.
- Possible control of the thermally driven (or the self-organized alpha pressure driven) bootstrap current needs to be evaluated.

ELM-FREE HIGH PERFORMANCE

Stellarators can operate in high-performance, high-confinement discharge modes both with and without ELMs. Edge localized modes are often observed in stellarator H-mode discharges, but the available data and understanding are more limited than in tokamaks at present. As in tokamaks, stellarator ELMs are associated with strong pressure gradients at the plasma edge, and act to flush out impurities. The appearance of stellarator ELMs depends sensitively on the edge rotational transform. An ELM-free, high-density H-mode attained in stellarators exhibits good thermal confinement with low impurity confinement times. Stellarator optimization may allow for good core confinement with edge conditions amenable to ELM suppression. The application of non-axisymmetric fields in the edge region has been shown in some cases to suppress ELMs in tokamaks.

Research Requirements

- Understand the mechanisms of enhanced transport in the edge thought to be responsible for ELM suppression with non-axisymmetric perturbations.
- Understand impurity transport in stellarator discharges without ELMs that show low impurity accumulation. Determine whether the ELM-free HDH mode can extrapolate to a burning plasma.
- High-performance QS plasmas are needed to determine whether lower edge pressure gradients are compatible with large pressure gradients in the core.

SUPERCONDUCTING STELLARATOR COILS

High T_c superconductors could have a large potential impact on fusion devices as identified in the FESAC "Priorities, Gaps, and Opportunities" report, and is a focus of Thrust 7. Second generation high-temperature superconductors (HTS) offer many advantages for fusion magnets over present technology. This is especially true for stellarators. The conductors can have small curvature radii, advantageous for stellarator coils. Demountable joints provide new possibilities for access and maintenance. Quench protection can be built into the magnet without large amounts of stabilizer. Higher current densities, critical field, and broader temperature range ease engineering design and cryo-loads. Novel manufacturing techniques are possible with the second-generation material, which might permit a direct application to an otherwise structural element. A staged program is needed to evaluate the suitability of HTS for stellarator coils or flux shaping components.

Research Requirements

Design and prototype work should be done so that, if advantageous, this technology could be applied to all fusion systems in the ITER-era.

- Undertake an engineering analysis of an example stellarator coil to determine critical elements in an R&D strategy for application of high T_c to stellarators.
- Carry out appropriate R&D through fabrication and testing of a model coil.

- Investigate how demountable coils could improve maintenance issues for stellarators.
- Determine whether conductor elements can be “printed” on structural elements for coil formation.

THE SPHERICAL TORUS

Introduction

The spherical torus (ST) is a low aspect ratio tokamak that offers advantageous physical properties due to very strong magnetic curvature and compact geometry (Figure 3). This configuration produces high plasma pressure relative to the external magnetic pressure and strongly affects plasma stability and confinement. It offers the promise of simple magnet design, reduced size, cost, and ease of maintainability. A description of the ST can be found in the Report of the FESAC Toroidal Alternates Panel (2008) (http://fusion.gat.com/tap/).

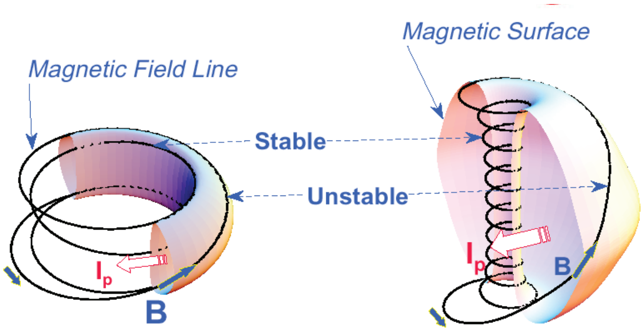

Figure 3. ST magnetic field geometry compared to conventional tokamak geometry.

Research Goals

The ReNeW ST panel largely adheres to the ITER-era goal stated in the executive summary of the TAP Report and the topical areas of research, with some important clarifications of emphasis and alterations. The ITER-era goal is stated as: *“Establish the ST knowledge base to be ready to construct a low aspect-ratio fusion component testing facility that provides high heat flux, neutron flux, and duty factor needed to inform the design of a demonstration fusion power plant.”* The ReNeW ST Panel clarifies that properly informing the design of a demonstration fusion power plant requires that research extend past the needs of an ST fusion nuclear science component testing facility (ST-CTF) in the ITER-era. It is equally important that this research examine a high level of plasma control flexibility and performance beyond baseline ST-CTF design needs to minimize performance risk for ST-CTF. This is because an ST-CTF will not be a research facility, but rather an *application* of ST research, with limited diagnostics and control variation capabilities to determine design optimizations. Other proposed ST applications are the driver for a fusion-fission hybrid device, and an ST-based demonstration fusion power plant (DEMO). The majority of the ST Panel further suggests that research aggressively pursue improvements to the ST concept that advance an ST-based DEMO, proposed by US and international researchers. These clarifications, related to minimizing the risk of attaining the ITER-era goal, were adopted when preparing the research needs for ST research described in this document. The original twelve critical research issues defined in the TAP report are retained, now expanding the “3-D fields” critical issue to more broadly include needed research in “Stability and Steady-State Control.”

Research Requirements

Key research needs to support the ST ITER-era goal divide into areas of: (1) plasma initiation and ramp-up, largely or entirely generated without a central solenoid due to ST space restrictions, (2) plasma-material interface, handling the uniquely high heat fluxes found in the inherently compact ST design, (3) understanding of energy transport — especially of the plasma electrons — which dominates in present high-performance ST devices, (4) stability and steady-state control for continuous operation at the high normalized pressure (beta) and with very broad current profiles of ST plasmas, at low levels of stored energy fluctuation, (5) technological development to support unique ST challenges including magnet technology, and (6) integration of the elements to demonstrate plasma sustainment and understand the interactions of the components for confident extrapolation to ST-CTF and DEMO. This collection forms a group of research thrust elements (see Thrust 16) with the twelve individual FESAC TAP critical issues included in these elements. Advancements in theory and modeling, critical to this plan, are integrated into the research requirements. No priority is implied by subject ordering.

ST AREA 1: PLASMA CURRENT INITIATION AND RAMP-UP

Plasma initiation and ramp-up techniques for the ST need to further advance by largely or entirely supporting this process without a central solenoid to generate plasma current.

INITIATION BY MAGNETIC HELICITY INJECTION

Research Requirements

Driving current parallel to externally applied magnetic fields, by applying a voltage to electrodes on the magnetic field lines, can generate plasma current by magnetic turbulence (a technique called magnetic helicity injection — HI). Research focusing on understanding the mechanisms that limit the achievable current and on determining the characteristics of the resulting plasma is needed. This requires dedicated experiments on STs with direct current HI capability (from localized “point” sources, and toroidal “coaxial” electrodes) and comparison to theory-based physical models. Experimental HI research is needed to develop local current plasma sources that provide very high current density, as well as toroidal metal electrodes that minimize plasma impurity content and simpler insulator configurations that are more easily adaptable to future STs. Testing coaxial configurations would also benefit point source HI.

Specific topics required for HI study include current, particle, energy, and momentum transport across magnetic fields in a turbulent system and the role of flows (due to strongly biasing the plasma edge) on the plasma equilibrium, stability, and transport as the plasma forms and evolves. Of particular interest is assessing the applicability of simple steady-state, two-dimensional models in describing an inherently three-dimensional dynamic system. Accurate predictive capability of the resulting plasma characteristics is needed to assess coupling to alternative current drive methods to further increase the plasma current to required levels. These validated models are needed to determine the required noninductive current drive capability of future ST designs.

Plasma current initiation via external magnets (outer poloidal field induction) in conjunction with HI and radiofrequency techniques can be assessed in existing and upgraded facilities. Initiation using a central solenoid is also possible by using either mineral insulated cable or retractable solenoid technology that both require engineering analysis to determine feasibility.

An initiation scenario using electromagnetic waves (electron Bernstein wave – EBW) needs to be developed and quantitatively compared to HI startup in the same facilities. Methods for understanding and controlling the plasma current and density evolution, the influence of plasma density fluctuations on the wave power coupling, and the levels of plasma current produced as a function of launched wave power need to be determined. Switching from an EBW-based startup to an alternate sustained heating and/or current drive technique must be shown.

Increasing the initially formed plasma current to desired levels requires additional techniques, including current driven by electromagnetic waves and/or energetic particles (e.g., neutral beam injection – NBI). Radiofrequency current drive is discussed in the next section. Neutral beam current drive (NBCD) for plasma current ramp-up is largely understood, but the effects of fast-particle instabilities on current drive efficiency must be determined using existing or upgraded facilities. Development of techniques to control electron transport during NBI current ramp-up is needed. Modeling should be performed to assess the impact on plasma stability of varying levels of angular momentum input for varying beam conditions. An assessment of the plasma current rate of rise compared to the plasma duration and the ability of the system to recover from disturbances that transiently decrease plasma current is required.

RADIOFREQUENCY HEATING AND CURRENT DRIVE

Many radiofrequency techniques may be applicable in higher toroidal field (1.5 – 3 tesla) STs, such as a CTF, to heat and/or drive current in the plasma. These include High Harmonic Fast Wave (HHFW) and EBW heating, electron cyclotron heating and current drive (ECH/ECCD), lower hybrid current drive (LHCD), mode conversion heating and current drive (MCH/MCCD), Alfvén wave heating or current drive (AWH/AWCD), as well as other heating and current drive schemes in the ion cyclotron range of frequencies (ICRF).

The requirements of future ST devices for radiofrequency heating and current drive need to be defined.

Research Requirements

The applicability of various radiofrequency heating and current drive scenarios to higher toroidal field STs, especially with significant alpha particle populations, has not been modeled with state-of-the-art radiofrequency codes. Present designs for an ST-CTF do not rely on radiofrequency heating and current drive to sustain the plasma. Needs in the plasma current initiation and ramp-up phase are not fully known. Needs for a DEMO operating at higher toroidal field and very high plasma pressure are not yet defined.

Central plasma heating and current drive (e.g., HHFW) is likely to be useful during both plasma initiation and ramp-up. The sufficiency of HHFW absorption must be determined for the range of plasma density and temperature in this period, as well as the operating frequency. The practical maximum operating frequency for high-power transmitters is presently ~ 120-140 MHz, corresponding to the fourth harmonic of the deuterium cyclotron frequency at a magnetic field of 2 Telsa. Optimal wave coupler design needs to be determined. Excitation of surface waves and radiofrequency sheath effects must be evaluated during the plasma initiation period.

Non-central current drive (including EBW, ECH/ECCD, and MCH/MCCD) have many possible uses to control the current profile, but no such technique has been demonstrated in an ST. Steering of ECCD may not be so straightforward in STs with high magnetic field. It is not presently known how to steer EBW, and MCH/MCCD is not steerable. Coupler design is an issue for all techniques. For MCH/MCCD, couplers in the ICRF range are large, and must be designed to tolerate large neutron fluxes when fusion reaction rates are high. The ICRF coupler design for ITER provides a starting point. For EBW, it is not presently clear that couplers can be effectively designed to function over a wide enough range of plasma edge density gradient and magnetic field line angle to be efficient throughout the plasma initiation and ramp-up period. Efficient coupling during constant plasma conditions is probably closer to realization. ST coupler design for ECH/ECCD can be heavily leveraged using conventional tokamak research. Simulations have indicated that EBWCD (Ohkawa current drive) may be quite efficient in the plasma edge region. LHCD is a candidate edge current drive technique. However, LHCD has never been seriously investigated in the ST. Modeling is needed at magnetic fields representative of an ST-CTF or ST-DEMO. Other research needs include modeling and higher-field ST experiments to validate theory.

ST AREA 2: PLASMA-MATERIAL INTERFACE

While the majority of plasma-material interface issues related to plasma-wall interactions, plasma facing components, and internal device components are common to all magnetic configurations, the TAP report identified several ST-specific knowledge gaps. The main goal for the ST research is the development of normal and off-normal heat and particle flux handling strategies compatible with plasma core and extreme edge plasma operating conditions, especially long plasma duration divertor and first wall power handling, particle control at low normalized plasma densities, and integration of these with high normalized plasma pressure and high plasma confinement operation. At present, two experimental facilities with graphite tiles facing the plasma (NSTX and MAST) contribute to the experimental understanding of these issues.

Research Requirements

1) Divertor and first wall power handling techniques: These need to be developed to reduce steady-state peak steady-state heat fluxes from the projected 20-60 MW/m^2 to 10 MW/m^2 or lower, and transient loads to 0.5 MJ/m^2 or lower in future ST-based devices with input power of tens of megawatts. This goal includes experimental and theoretical understanding of extreme edge plasma (scrape-off layer — SOL) and divertor plasma electron and ion heat transport to develop predictive modeling and scalings for peak flux and scrape-off layer flux width for projections to future devices. The roles of conduction, convective (turbulent) transport, radiation transport, particle flow to the divertor and first wall at low plasma collisionality need to be clarified. At low SOL density, kinetic effects will play a larger role in heat transport. New numerical analysis with gyro-fluid and gyrokinetic transport and turbulence models will be required. To help validate the models, new diagnostic measurements to elucidate the edge electron and ion energy distribution

functions, edge flows, and edge neutral density will be needed. Divertor and first wall heat mitigation techniques at low collisionality will have to be developed. The collection of divertor power reduction techniques, e.g., innovative magnetic divertor configurations (the X, Super-X and "Snowflake" divertors), volumetric momentum and power exhaust, and innovative plasma facing components, e.g., liquid metals or moving "pebbles," should be tested in the unique, high heat flux, low scrape-off layer collisionality operational regime, and within limitations imposed by the ST magnetic geometry. Initial evaluation of these techniques can be performed in upgraded facilities; however, new scaled ST devices (with high atomic number metal walls, or porous liquid metal walls) will be required for an integrated performance demonstration.

2) Particle control techniques: Efficient pumping techniques that would enable low normalized density operation compatible with proposed heat mitigation solutions and high core and edge plasma performance must be demonstrated. For impurity and helium density control, and for hydrogenic density reduction and control, the candidate techniques include cryopumps, and low recycling liquid metal walls and divertor. A large number of engineering issues for cryopumps will have to be addressed including tritium retention, cryopump operation and regeneration during continuous plasma operation, and material issues. While the concept of low recycling of particles at the plasma edge by using liquid metal walls shows promise, a number of issues (e.g., its compatibility with high-performance steady-state plasma operation, helium pumping, and off-normal event handling) must be solved. In addition to pumping tools, efficient fueling techniques must be developed. Fueling techniques include central NBI fueling, pellet injection, compact toroid injection, and supersonic high-density gas injection. However, their compatibility with low normalized density continuous plasma operation must be demonstrated.

3) Integration of particle and power reduction techniques: This integration with a high-performance core plasma is a critical research need for the ST. The plasma edge region acts as an interface between the SOL and core plasma, thus the impact of heat and particle control techniques on plasma edge stability and confinement may be an issue.

A number of plasma-material interface issues common to conventional tokamaks must also be solved. Critical issues include plasma facing component design (shape, geometry, engineering, materials). Erosion of plasma facing component materials, impurity generation and transport, and tritium retention in these components will have to be solved.

While some parts of this research can be addressed in near-term upgraded STs, other toroidal and laboratory experiments, and dedicated modeling efforts, the ultimate demonstration of an integrated plasma-material interface solution must be shown in a scaled ST configuration.

ST AREA 3: ENERGY TRANSPORT

ELECTRON ENERGY TRANSPORT

Experiments indicate that the ST plasma may have unique energy transport characteristics, related to rapid electron thermal transport being the main energy loss mechanism in high normalized pressure (beta), high thermal confinement, NBI-heated conditions. The knowledge gap is insufficient understanding of electron thermal transport at low plasma aspect ratio, high beta, and low plasma collisionality to reliably predict confinement for an ST-CTF or DEMO.

Research Requirements

Electron transport studies over an extended range of key plasma parameters: Plasma electron temperature gradient (ETG) and micro-tearing of the magnetic field are predicted to be the dominant instabilities in the electron temperature, T_e, gradient region of NBI-heated, high-confinement ST plasmas. Trapped electron instabilities might also contribute at lower plasma electron collisionality, ν_e. Since the plasma temperature and collisionality scaling of transport induced by these instabilities are very different, it is essential to establish which contributes most, locally inside the plasma. Another important question is if ETG and micro-tearing instability-induced transport can be reduced by the large electric and magnetic field (ExB) shear possible in the ST. Finally, to inform ITER and DEMO, it will be important to investigate turbulence suppression mechanisms that do not need external rotation input, such as predicted pressure gradient effects. Addressing these questions requires transport scaling studies over an extended range of T_e, ν_e, ExB shear, and pressure gradient, compared to present experiments (e.g., at more than twice the present T_e, and one-tenth of ν_e). This is achievable in upgraded ST devices, or a performance extension experiment (see TAP report). Study of edge plasma electron thermal transport with low recycling lithium walls and in the presence of magnetic fields used for active control of plasma instabilities is also needed for STs and other toroidal devices.

Diagnostics for high and low wavelength, electrostatic and magnetic turbulence in STs: To understand the role of ETG and micro-tearing instabilities in ST electron transport one needs to investigate the existence of ion-gyroradius (ρ_i) scale ETG "streamers" and magnetic islands. While the existence of ETG streamers is an important question for all magnetically confined devices, the low magnetic field, large ρ_i scale of ST plasmas make them favorable for these studies. This research will require the development of novel and improved diagnostic techniques. (See ReNeW white papers by E. Mazzucato, and by K. Tritz, et al.).

Study of electron transport driven by fast ion instabilities: A correlation between central electron transport and fast ion driven instabilities (global Alfvén eigenmodes — GAEs) has recently been found in NSTX, with the instabilities driving stochastic transport at diffusivities greater than $10m^2/s$. This phenomenon could have an impact on ST-CTF performance and might prove relevant for any burning plasma. It also could have beneficial uses, such as T_e profile control or current drive in an ST-DEMO. For this study, specialized magnetic and density fluctuation diagnostics are needed, with a large field of view of the plasma, moderate resolution (a few cm), very high speed (several MHz) and signal-to-noise ratio ($>10^3$). Theoretical models capable of predicting the GAE characteristics and the associated transport under different T_e, collisionality, q-profile, magnetic and flow shear conditions, as well as the interaction of this type of transport with conventional gradient-driven transport, are needed. Methods to control this transport, for example using localized radiofrequency suppression and excitation of the GAEs, also need to be investigated.

Development of predictive capability for anomalous electron transport at high β and low ν_e: This capability is needed to optimize the performance of future ST devices, including the determination of optimal aspect ratio, and to control plasma profiles. Research needs to develop this capability include: (i) development and validation of theoretical models for electromagnetic turbulence at high β and low ν_e, including modeled diagnostic output for comparison with fluctuation measurements, and (ii) improved transport measurements. There seem to be multiple physical effects driving stochastic heating and energy loss in the ST. Measurements of the energy distribution function for all particles are therefore needed to study the physical mechanisms. In addition, tools that will enable perturbative electron transport measurements in the ST, such as EBW heating and fast electron temperature diagnostics, are highly desirable, and (iii) development of electron transport control tools, such as localized electron heating, magnetic field line pitch (*q*-profile) reversal, flow shear, and control of fast ion instabilities.

ION SCALE TRANSPORT

In strongly rotating high-confinement ST plasmas, ion thermal and particle transport is near neoclassical levels. There is insufficient understanding of long-wavelength turbulence suppression mechanisms in the ST to allow predictions for CTF or DEMO.

Research Requirements

Most needs overlap with those for electron transport. Ion specific requirements are:

Study of the role of flow shear, magnetic shear, pressure gradient, zonal flows and poloidal rotation over an extended parameter range: While ExB shear is believed to be responsible for the neoclassical ion transport observed in strongly rotating ST plasmas, a direct experimental link has yet to be made. The amount of ExB turbulence suppression is unknown. Observed anomalous momentum transport suggests “remnants” of long-wavelength turbulence. There is a need to investigate if zonal flows can be generated by electron-scale turbulence, their inherent *q*-dependence, and their relation to scrape-off layer flows.

Study of transport regimes with low rotation input: To inform DEMO, ion transport needs to be investigated in regimes where pressure-driven flows and/or pressure gradients suppress high-wavelength turbulence. The heating power required to produce high confinement needs study.

Study of the role of neoclassical impurity transport at low magnetic field: While neoclassical particle transport is desirable for an ST-CTF or DEMO, there could be deleterious effects of neoclassical high atomic number impurity transport at large gyroradius, such as collisional penetration of edge density gradients, or strong impurity accumulation.

Study of the role of anomalous ion heating: Recent experimental results suggest that fast ion driven instabilities may stochastically heat ions. Intense micro-tearing might also have a similar effect through small-scale magnetic reconnections.

ST AREA 4: STABILITY AND STEADY-STATE CONTROL

CONTROL OF INSTABILITIES

The broad plasma current profiles and near-spherical geometry of the ST have a strong impact on stability. The ability to maintain continuous high normalized plasma pressure (beta) operation required for ST-CTF and DEMO, at required low levels of plasma stored energy fluctuation and disruption, has not been demonstrated, and may require innovative and pioneering solutions. The understanding needed to confidently extrapolate such operation to future devices must be acquired by targeted research at reduced plasma collisionality. Greater constraints on plasma control flexibility in ST-CTF and DEMO require solutions to be optimized in preceding devices.

Research Requirements

Reducing stored energy fluctuation and disruption probability: Plasma instabilities are a significant cause of stored energy/plasma current fluctuations and plasma termination events called major disruptions. Understanding the cause for instability allows fluctuation/disruption control or avoidance and is considered below. Further considerations are discussed later in this Chapter.

Kink/ballooning modes: The ST operates in a uniquely low range of plasma internal inductance, l_i (approximately 0.35 for ST-CTF, lower for ST-DEMO) and will make it more susceptible to the current-driven kink instability, which by definition is unstable at any value of the stability parameter, normalized beta, β_N ($\beta_N \equiv 10^8 \beta_t aB/I_p$ where a, B, and I_p are the plasma minor radius, magnetic field, and current). It is crucial to understand and characterize the robustness of low l_i stability to variations in current, pressure and plasma rotation profiles, and collisionality.

Resistive wall modes (RWMs): These instabilities lead to plasma termination by disruption, but can be stabilized or actively controlled. The ratio of the normalized pressure that can be maintained stably to the plasma internal inductance (β_N/l_i) is an important stability parameter with broad current profiles for the pressure-driven kink instability and RWMs. The low l_i of the ST yields a very high ratio of β_N/l_i (exceeding 17 in ST-CTF). Present experiments approach this stability level, and this operational regime may be accessible with RWM stabilization via plasma rotation and active control, but present experiments show significant stored energy fluctuation and disruption probability in these plasmas. Kinetic theory and fast particle effects are important to determine RWM stability, and reveal that the instability can occur at relatively high rotation speeds, far greater than past projections of a “critical rotation” speed. These “weak rotation profiles” are theoretically investigated by kinetic evaluation of RWM stability, and continued comparison between theory and experiment is needed. Along with β_N, l_i, plasma rotation, V_ϕ and fast particle pressure profiles, plasma collisionality is an important variable determining stability, so a significant decrease (by a factor of 10) in this parameter toward ST-CTF levels is needed.

Edge localized modes (ELMs): Edge localized modes are instabilities that generate explosive fluctuations in the stored kinetic energy in the plasma edge region, manifesting as the rapid expulsion of hot dense plasma, which produces impulsive heat loads on divertor plates and other plasma facing components. ELMs can also affect plasma rotation and trigger other deleterious modes, including RWMs. With relatively low fields, high-current densities, and broad current profiles, ST plasmas could allow excitation of the underlying instabilities under a wider range of plasma conditions than conventional tokamaks. As such, ST plasmas may offer a unique opportunity to rigorously test theory with comparisons to detailed experimental data. To produce results that are applicable to conditions expected in ST-CTF discharges, these studies will need to address the role of plasma collisionality, with plasma flows and equilibrium profiles consistent with global mode stability.

Neoclassical tearing modes (NTMs): See later section addressing NTM physics and control.

Control considerations and innovations: Significant reduction of disruption probability requires control innovations that need to be tested in the ITER era. Control of instability amplitude, plasma pressure, plasma rotation, and magnetic field pitch angle (q) profile should be used to eliminate disruptions with high reliability. Feedback control of multiple instabilities possible in high-beta ST plasmas needs to be addressed using upgraded 3-D control fields and advanced control algorithms. Non-magnetic instability sensors should be tested. The addition of moderate 3-D shaping from applied 3-D control fields should be studied to minimize disruption occurrence. Real-time assessment of plasma stability based on measured stable RWM behavior and real-time theoretical stability computation should be considered. Transient reduction in plasma energy transport that increases plasma pressure and creates global instability needs to be controlled by heating or plasma energy confinement control. Plasma rotation needs to be controlled to avoid profiles leading to unstable RWMs, or rotation reduction caused by stable RWMs, saturated NTMs, ELMs, and error fields. Plasma pressure control is possible via localized heating, fueling, and transport changes, and q control is possible with localized current drive. Control of plasma rotation and rotation shear is possible by applied 3-D magnetic fields, but depends on plasma collisionality. This theoretical dependence needs to be experimentally verified at reduced collisionality approaching ST-CTF levels. Control systems considered should be compatible with the continuous operation and high neutron fluence anticipated for ST-CTF.

DISRUPTIONS

Disruptions terminate plasma operation and must be avoided. Present ST experiments have not achieved the integrated plasma performance conditions needed for an ST-based CTF or reactor, and these conditions could strongly influence the nature and frequency of disruptions. Disruption avoidance for durations progressively up to about five to six orders of magnitude longer than present ST plasma pulse durations and progressively up to about three orders of magnitude longer than present or planned long-pulse devices (including ITER) will be required.

Research Requirements

To sustain stable operation near (for baseline ST-CTF operation) and above (for improved ST-CTF fusion performance and DEMO) stability limits for non-rotating plasmas, CTF-relevant plasma shaping and plasma profile/RWM/ELM control models, actuators, and diagnostics should be extensively tested for reliable disruption avoidance in present facilities. The frequency of disruptions should be quantified as a function of proximity to the key stability limits to assess the potential trade-off between higher fusion performance and reduced disruption probability. Additional design and engineering activities are needed to determine the allowable frequency and magnitude of disruptions for various ST applications based on allowable damage limits to internal components from electromagnetic and thermal loads and high-energy electrons. It is noted that 10^6 seconds of continuous plasma operation is the long-term objective for the component testing mission of a CTF, and thus CTF will likely require disruption occurrence rates of less than $10^{-6}s^{-1}$ (a DEMO reactor would likely require one to two orders of magnitude lower disruption frequency). To make substantial progress toward the disruption avoidance requirements of future fusion devices, ST pulse durations should be extended by one to three orders of magnitude beyond present capabilities. This implies that the present 1s plasma duration should be extended to 10^1-10^3s, and ~10^3s/year of accumulated operating time should be extended to 10^4-10^6s/year with a substantial fraction of that time achieved free of major disruptions.

ENERGETIC PARTICLE INSTABILITIES

The ReNeW ST Panel extends the TAP knowledge gap statement for this area to be "Impact of energetic particle instabilities on neutral beam current drive, heating, electron transport, and alpha channeling in the ST" to include important energetic particle (EP) instability effects on electron transport, as well as so-called alpha particle channeling issues.

Research Requirements

A unique feature of STs is that energetic particles created in the plasma by external heating such as NBI and ICRH, or by fusion alpha particles in an ST reactor, have velocities far greater than the velocity of transverse magnetic waves in the plasma (super-Alfvénic particles). This produces a rich variety of EP instabilities. High velocity (fast) ions can effectively interact with various collective phenomena in the plasma, notably low frequency (Alfvén) modes capable of inducing energetic particle radial transport. Instabilities with high frequencies (below, or near the ion cyclotron frequency) can change particle energy and be used to transfer the energetic particle energy directly to plasma thermal ions, thereby avoiding heating of plasma electrons. High-frequency modes can resonantly interact with plasma thermal electrons, which is a newly discovered phenomenon that can pose a great challenge for ST design.

Energetic particle anomalous transport effects on heating and current drive: Experiments in recent years on conventional tokamaks and STs demonstrated large losses of fast ions due to the Alfvénic instabilities — in some discharges up to 40% of the heating NBI ions, affecting heating efficiency and the current drive. These losses are poorly understood theoretically. The following steps are needed to close the knowledge gap:

- Develop numerical and theoretical models for EP transport predictions in the presence of multiple instabilities expected in burning plasmas and ITER: This should include linear theory predictions for Alfvén instability-free regimes and nonlinear simulations for tolerable Alfvén instability activity.

- Develop new diagnostics for plasma oscillation and EP profile measurements: This is needed to establish an experimental database of EP losses and wall heat loads to help verification and validation and foster reliable projections to next-step ST devices.
- Develop active control tools for linear and nonlinear EP transport control: Notably in the nonlinear regime, control over hole-clump generation is desirable.

Effect of energetic particle driven modes on thermal plasma: The electron transport induced by energetic particle driven instabilities such as global Alfvén eigenmodes is unknown, and energetic particle energy channeling in which fast ion-induced multiple compressional Alfvén eigenmodes (CAEs) transfer their energy from the EP source to thermal ions by a transient resonance is not well understood. The following studies are needed to fill the knowledge gaps:

- Conduct nonlinear theoretical studies of multiple CAE, GAE and energetic particle interactions to understand their effect on fast ions.
- Develop a program to study these effects experimentally by measuring internal mode structures and polarization, and the effects on EPs on a fast time scale during GAE and CAE bursts (on the order of 1 msec and faster).
- Engage in active antenna studies of wave-particle interaction with external excitation of high-frequency modes.

NEOCLASSICAL TEARING MODES

Neoclassical tearing modes are prominent in high-performance tokamak and ST operation, and are among many instabilities that can reduce confinement and/or cause disruptions. For steady-state operation, a large fraction of current self-generated by the plasma (bootstrap current) is clearly desirable. Since this current is produced by the plasma pressure gradient, it exposes the pressure gradient free-energy as a mechanism to destabilize macroscopic tearing instabilities, with associated magnetic island formation. For high-performance ST operation, it is highly desirable to avoid neoclassical tearing modes in rotating plasmas near the ideal plasma pressure-driven stability limits.

An extensive database is available from conventional tokamak experiments describing NTM instability threshold, formation, and saturation processes through a combination of analytic modeling and experimental investigation. Largely, different elements of the theory can be parameterized by a set of dimensionless numbers, normalized gyroradius, measures of plasma toroidal rotation and toroidal rotation shear, collisionality, etc., from which extrapolations and predictions of future devices are predicted. Present ST operation feeds into this physics program and helps firmly establish the relevant physics. Since STs operate in the extremes of normalized rotation shear, plasma shaping, etc., it may be possible to identify a desirable operational region that makes STs somewhat less sensitive to the deleterious effects of NTMs. A comprehensive theory describing all of NTM physics remains elusive, and by leveraging a unique operational space, ST experiments make a novel contribution in the development of theory.

Research Requirements

There are two principal approaches for dealing with NTMs in tokamaks. The first is NTM avoidance by operating with q profiles that prevent low order resonances, created at specific depths in the plasma based on the magnetic field line pitch. The second is to use active feedback by employing localized ECCD to stabilize the slowly growing NTMs. In present STs, NTM avoidance by operating with an elevated value of the minimum q, q_{min} is the preferred method. However, present experiments are largely unable to sustain high-performance discharges with elevated q_{min} and hence, NTM instabilities remain a prominent issue. Current drive mechanisms that will maintain q_{min} greater than two are highly desired in future STs. Active feedback with ECCD is not available in present-day STs due to wave penetration issues related to the relatively low magnetic field strength. Success in mitigating or eliminating NTMs has been shown in NSTX experiments during the past two years, related to the application of lithium to the inside wall of the device. The physical mechanism responsible for this effect should be conclusively determined and understood to extrapolate this result to future STs.

Theoretical models do not yet exist to accurately predict all of the trends in the experimental data. Of particular note to the ST is the scaling with normalized ion gyroradius, ρ^*, and with large plasma rotation and shear. As such, enhanced emphasis is needed on ST-relevant theory and computation in this area. Coupled with this study is the need to develop diagnostics that can probe the detailed structure of magnetic island physics in STs.

Tools to address NTM physics and stabilization need to be developed. Upgrading present STs to higher magnetic field and current may provide a mechanism to viable scenarios that avoid NTMs. Present methods to control plasma current and rotation profiles need to be further expanded. The use of localized current drive sources for active control of NTMs in STs needs to be assessed. In particular, localized ECCD should be explored as a possibility at higher magnetic field. Other options should be explored for localized current drive in STs that may provide a mechanism to stabilize NTMs.

ST AREA 5: TECHNOLOGY FOR STEADY STATE

MAGNETS

The tight space and inability to fully neutron-shield the center section of an ST motivates elimination or simplification of electromagnets in this region.

Research Requirements

The simplest approach to the toroidal field magnet has a single-turn, resistive, intensely cooled copper centerpost. No single-turn centerpost ST has been operated anywhere in the world since the START device. Spherical torus fusion reactor design has focused on the single-turn centerpost because of the difficulty in finding an insulator, which will not become conductive due to neutron damage in the presence of an electric field. Very low impedance power supplies, possibly homopolar generators, will be required to power the single-turn toroidal field magnet system, but have not been demonstrated within an order of magnitude of the currents required. Any maintainable system requires demountable joints between the centerpost and the outer legs of the toroidal field coils, to permit replacement of the centerpost. Demountable joints at the required current and current density have not been designed or demonstrated. The use of resistive shims in the center stack, or a centerpost construction with coaxial, air-gapped conductors, may permit designs with a modest number of turns (~ one per toroidal field [TF] leg). Superconducting TF designs are at very early stages. Recently, designs for high-temperature superconducting centerposts, shielded with more than 12 cm of tungsten to reduce neutron heating to an acceptable level, have been advanced, primarily in connection with fusion-fission hybrid device designs. Possible approaches to incorporate a central solenoid in a neutron environment have been advanced, but not fully investigated. These approaches include the installation of resistive shims in the TF centerpost to generate a solenoidal field, a retractable central solenoid, the use of an iron core magnetized by coils outside of the radiation zone, and the use of a mineral-insulated conductor to construct the solenoid. None of these approaches to a reactor-relevant central solenoid has been tested.

Degradation of the conductivity and mechanical strength (embrittlement) of the copper conductor in the neutron field must be studied. Optimization of the cooling channel design must include variability of resistive and nuclear heating, erosion, etc., within the centerpost and at channel surfaces, and manufacturing considerations. Tritium migration into the coolant must be quantified. Demountable joint integrity requires engineering development and innovation. The design for centerpost demounting, remote handling of neutron-activated joint disassembly, and centerpost replacement must be addressed. Power supply development is required — multi-megampere power supply requirements are unprecedented and challenging. Fault modes of very high current power systems must be considered. The primary needs for superconducting centerposts are shielding design, quantification of neutron heating, and reduction in the tritium breeding fraction. Design issues for the central solenoid are whether to use removable, resistively shimmed, mineral insulated, or iron-core approaches. The mechanical integrity of mineral insulation is an issue. Startup scenario development for the resistively shimmed approach is required. Radiation qualification and testing of all designs is clearly needed.

CONTINUOUS NBI SYSTEMS

Traditional methods of NBI implementation and operation are incompatible with steady-state operation and need to be replaced by new techniques.

Research Requirements

The desired beam energy and total power of future NBI systems will depend on the steady-state plasma characteristics. An assessment of the required beam energy must consider the total heating power required. At lower beam energy, the beam power density may be insufficient to maintain the desired total power input. Positive ion sources are not likely to be viable, so negative ion sources must be capable of achieving the required beam parameters. Cathode filament lifetime in present negative ion beam sources can be extended by research to develop better arc detection and arresting. Research is required to assess the advantages of radiofrequency ion sources for potential development, thereby eliminating the need for filaments. Indirectly heated cathode technology could also be developed into a negative ion source. This technology has been used in positive ion sources without arcing, but requires further research for use in negative ion sources. The ion source requirements for ITER are equivalent to those needed for steady-state, so research in support of ITER will develop the needed technology for negative ion beam sources.

Research is needed to develop methods of handling or reducing the high neutral gas load that results from conventional gas neutralization. The feasibility of using continuous, high-throughput cryopumps to allow conventional gas neutralization needs to be evaluated. Conventional gas neutralization has uniform target thickness over the transverse cross section of the ion beam, which is a consideration for other neutralization schemes.

Photo-detachment neutralization requires the development of lasers capable of continuous, high-efficiency operation and the ability to withstand neutron damage. Such a system would be energy intensive, so an engineering assessment of the benefits must be performed. Plasma neutralization requires the development of a plasma source with high ionization fraction and uniformity across the beam cross section. This technique must be evaluated as it may not significantly reduce the gas loading. Lithium vapor jet neutralization requires a study of the beam neutralization physics as well as research on the lithium handling hardware. The lithium recovery system and a means to control the impurity content in the lithium loop would need to be developed.

ST AREA 6: INTEGRATION AT HIGH BETA

Favorable operating scenarios for an ST-based component test facility are projected to have low normalized density fraction (~20-30%) and low core plasma electron collisionality $\nu_e^* \sim 10^{-3}$, utilize up to 50% beam-driven current fraction, operate in a plasma with ions hotter than electrons, and with high-energy confinement operation at enhancement factors up to 50% above the predictions for ITER. Present ST experiments may have difficulty achieving normalized density fractions below 50% and have $\nu_e^* > 10^{-2} – 10^{-1}$, have sustained only 10-15% neutral beam-driven plasma current fraction (due to high-density operation and non-optimal beam injection geometry), and have sustained confinement enhancements ~10-15% above ITER predictions. Potentially more favorable energy transport mechanisms for the electrons and ions have been indicated in present-day ST experiments — accentuating the importance of understanding energy transport in the ST — especially at reduced collisionality. These core plasma parameters have not yet been sustained for long plasma durations, i.e., for many current redistribution, pumping, or wall/divertor equilibration times.

Research Requirements

Based on plasma thermal confinement predictions observed thus far in NSTX and MAST, a factor of 2 to 4 increase in toroidal magnetic field and plasma current is needed to access increased plasma temperature and reduced plasma collisionality to values approaching (to within a factor of ~5) those projected for an ST-CTF or an ST DEMO. Improved density control is needed to sustain reduced collisionality, and density pumping by liquid lithium plasma facing components and/or divertor cryopump systems should be tested to assess the accessibility of a low normalized density fraction in ST-CTF-relevant integrated scenarios. Sufficiently high $\beta_t \geq$ ~15% at reduced collisionality is needed to develop an understanding of the underlying causes and scalings of electron (and ion) transport in high normalized pressure (high-β) ST regimes and in compact high-β systems generally. Enhanced gyrokinetic turbulence simulation capabilities are needed to understand the relative roles of long-to-intermediate wavelength versus short wavelength turbulence in causing electron transport. Given the very high plasma energy confinement potentially needed for the ST-CTF application, very low recycling regimes should be tested on present and near-term experiments to assess predictions of improved confinement at the reduced toroidal magnetic field and high-β of the ST.

Increased NBI current drive efficiency and control is needed to test fully noninductive operation of an ST as needed for an ST-CTF. Reduced collisionality, increased NBI heating power, and injection geometries more favorable for NBI current drive and current profile control are needed to assess the predicted NBI current drive capabilities. These additional capabilities are especially important for testing ramp-up of the plasma current that is assumed to be achieved using NBI heating and current drive in an ST-CTF. ST operating scenarios could have a substantial population of super-Alfvénic ions and associated energetic particle instabilities. Since these instabilities can result in fast ion transport — which is not presently well understood — improved diagnosis and theory and computation of fast ion redistribution by Alfvénic activity is needed for reliably projecting the performance of future STs. Sufficient plasma duration (~10^1 to 10^2s) is required to achieve full plasma current profile relaxation following plasma ramp-up and during the sustainment phase, and to demonstrate profile controllability.

Exhaust solutions for high-particle flux and CTF-relevant high-heat flux must also be developed and assessed for compatibility with a sustained, fully noninductive, integrated high-performance core plasma. Such solutions might include liquid metal plasma facing components and/or high flux expansion and increased scrape-off layer magnetic field line-length as embodied in the X and Super-X divertor configurations. The above core and edge plasma integration must ultimately be achieved using the first-wall materials and operating conditions expected in an ST-CTF and/or reactor to ensure that acceptably low tritium retention can be achieved and that the first wall can tolerate infrequent off-normal events such as ELMs and disruptions.

Finally, new computational and theoretical capabilities supporting comprehensive time-dependent integrated modeling of the plasma evolution, using reduced models for core and edge transport and stability, are needed to develop and understand ST integrated scenarios, and to reliably project to the performance needed for an ST-CTF and beyond.

ST Available Means for Research and Actions to Address Research Needs

Present US and large international ST research facilities include (in order of decreasing plasma current capability) the NSTX (1 MA), PEGASUS, and LTX in the US, MAST (1 MA) in the U.K., and QUEST in Japan. Many smaller devices are operating internationally (see FESAC TAP report for greater detail). Proposed upgrades for these devices that have gone through initial review include NSTX-U and MAST-U (both 2 MA). These upgrades are critical to pursuing the ITER-era ST goal, but they and further upgrades may not be sufficient to fulfill this goal. Defining additional steps to resolve the ST physics issues described above and advance to an ST-CTF is evolving through the ReNeW process and beyond. The fusion nuclear science and technology actions for a CTF are addressed in Thrust 13.

The actions needed to address the ST research requirements stated in this Chapter are given in Thrust 16, "Developing the spherical torus to advance fusion nuclear science." A schematic timeline for this research is shown in Figure 4. Specific requirements for the indicated "ST Physics Validation at Long Pulse and High Field" step shown in the figure will be determined by research and design studies conducted in the coming years. A summary of the research requirements for all ST physics research areas is given in Table 1. This research and design analysis addresses any remaining issues that must be resolved to move the ST toward the burning plasma regime in support of fusion science research.

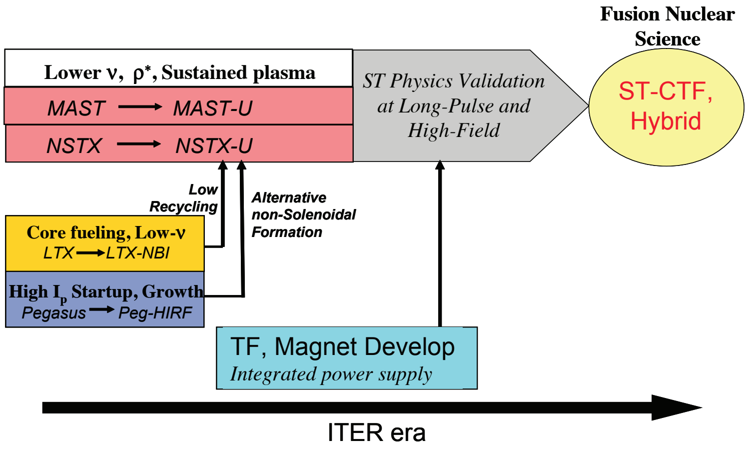

Figure 4. Timeline of ST Research leading to the ITER-era goal.

		Research Requirements		
	ST Scientific Issue	Required studies, advanced diagnostics, theoretical modeling	Major upgrades of existing facilities	ST physics validation at long-pulse and high-field
Area 1	**Plasma current initiation and ramp-up** under ST space constraints	Initiation: increase current to 4x present level in experiments, supported by validated, non-linear 3-D simulations.	Examine initiation current scaling at twice B_t. Double the NBI power and increase plasma duration 5x to produce ~ 1 MA plasma current.	ST device with I_p ramp-up systems needed for goal: $B_t > 2T$, $P_{NBI} \sim 30$ MW, sufficient pulse length to reach full I_p.
	Radio-frequency heating and current drive at ST magnetic field and density	Scoping of RF heating/CD at higher B_t, with alpha particles; improved modeling of RF edge effects for HHFW and EBW.	Investigate high B_t RF scenarios. Develop improved wave launchers for resilience to edge effects for high-power RF application.	RF scenario, launcher testing in long-pulse ST-goal plasma conditions, compatible with edge plasma, for heating/CD needs.
Area 2	**Plasma material interface** in compact ST geometry	Understanding of edge plasma, recycling, divertor heat flux width and scaling. Scoping studies of ST-level power handling techniques.	Implement advanced divertors (magnetic configs./cryopump, lithium wall) to reduce edge density, study high heat flux and particle control for longer duration.	Validate divertor/particle control physics in plasma approaching ST-CTF level: P/S and/or P/R near/at CTF levels, low normalized density edge.
Area 3	**Electron energy transport** **Ion scale transport** to understand unique ST confinement trends	Examine ETG and micro-tearing physics with short to long wavelength, electrostatic and electromagnetic turbulence diagnostics and non-linear gyrokinetic simulations.	Double B_t to 1T, I_p to 2 MA, double P_{NBI} to study ST electron thermal transport, ion energy, momentum, particle transport, and assess role of ion turbulence at reduced plasma collisionality (~ 1/10th), high beta, reduced $V\phi$.	Validate electron transport physics (including fast ion instabilities) in ST-CTF level plasma with margin: $B_t >$ 2T, $I_p > 4$ MA, $P_{NBI} \sim 10\text{-}30$ MW, T_e and T_i up to 10 keV, $\beta_t \sim 10\text{-}30\%$, varied $V\phi$.
Area 4	**Stability and steady-state control** with near-spherical geometry, broad current profiles, high β	First-principles understanding of RWM stabilization by $V\phi$, ν_i, P_{fast}; RMP suppression of ELMs; field-induced flow damping at low ν_i; control yielding insignificant fusion power fluctuation, non-magnetic sensors.	Double B_t to 1T, I_p to 2 MA, upgrade 3-D control fields, NBICD profile, fueling to study kink stability at low l_i, RWM stability at low l_i, ν_i, $V\phi$ and q profile, ELM mitigation at high edge β.	ST device with current, p, $V\phi$ profile, 3-D field, mode control for ST-goal plasma with margin: $\nu_{i,}$ ρ^* near/at CTF levels, $q_{min} = 1\text{-}3$, $l_i \sim 0.25\text{-}0.6$, $\beta_t \sim 10\text{-}30\%$, β_N up to ideal wall limit, varied $V\phi$.
	Disruptions: avoidance and mitigation for reliable continuous operation	Physics understanding of disruptions and underlying ST instabilities; real-time stability limit/margin calculations; advanced control components to avoid limits.	Demonstrate/understand disruption avoidance in non-inductively sustained high beta, low ν_i plasmas; assess disruption thresholds, margins for up to 10s.	Long pulse facility demonstrating disruption avoidance in integrated high-β plasmas relevant to ST goal - including goal-relevant error field, ELM, RWM control.
	Energetic particle (EP) instabilities: impact of super-Alfvenic ion-driven instabilities in ST geometry	Understanding of EP modes on NBCD, heating, electron transport, alpha channeling; non-linear 3D hybrid kinetic modeling; improve EP profile/fluctuation measurements.	Double B_t to 1T, I_p to 2 MA, with density control, greater off-axis-NBI to create P_{fast} distribution function relevant to ST-goal plasmas.	Study multiple EP instabilities in ST-goal plasma conditions. Establish limitations on plasma performance from multiple high frequency EP modes.
	Neoclassical tearing modes examining aspect ratio effects on stability	Understanding/validation of aspect ratio dependence of NTM stability theory; validated non-linear MHD calculations.	Double B_t to 1T, I_p to 2 MA, upgrade NBICD flexibility to sustain plasma with q profile elevated > 2 to test NTM avoidance at low A.	Implement required NBICD control system in long-pulse, high-field device to avoid NTM by elevated q, or other means (e.g. ECCD).
Area 5	**Magnets** for low-A device applications	Engineering studies for single-turn centerpost TF magnets; radiation-tolerant OH systems.	Double B_t to 1T to inform engineering design of ST goal-level device with $B_t >$ 2T	Device to prototype single-turn demountable TF, radiation-tolerant or removable central solenoid
	Continuous NBI	Engineering studies of long-pulse NBI application; RF ion sources.		Beam neutralization development for ST-CTF pulse lengths.
Area 6	**Integration at high beta** of ST physics needs	Integrated modeling of low normalized density, low collisionality plasma with high NBICD fraction.	Double B_t for low collisionality, low normalized density, increase NBICD to produce fully non-inductive sustained plasma	ST device with control and sustainment flexibility to determine optimal high performance ST-CTF plasma.

Table 1.

THE REVERSED FIELD PINCH

Introduction

During the ITER era, the goal for Reversed Field Pinch (RFP) research as identified by the FESAC Toroidal Alternates Panel (TAP) is to:

Establish the basis for a burning plasma experiment by developing an attractive self-consistent integrated scenario: favorable confinement in a sustained high-beta plasma with resistive wall stabilization.

The distinctive feature of the RFP configuration for its fusion application is that electrical current flowing in the plasma generates most of the confining magnetic field. The externally applied toroidal magnetic field is relatively small, about two orders of magnitude smaller than that in a tokamak with similar plasma current. This reduced external magnetic field requirement offers significant advantages for fusion application. For example, the RFP attains very high engineering beta (the ratio of the plasma pressure to the maximum magnetic field pressure at the magnets). Smaller magnets are allowed, which could be constructed using normal conductors (nonsuperconducting). Also, the plasma current provides large ohmic heating, reducing or perhaps eliminating requirements for complicated auxiliary plasma heating systems. While reduced external field underlies these potential advantages, it also leads to the primary scientific challenges for the RFP. Magnetic fluctuations arise more easily without a large externally imposed stabilizing field, and this has an impact on both plasma stability and confinement quality. Also, the steady-state sustainment of the RFP's relatively large plasma current is challenging.

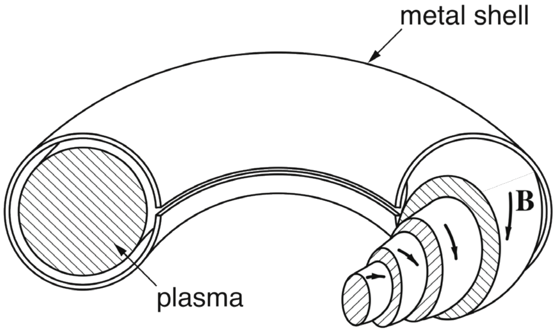

Figure 5. The RFP magnetic configuration. The direction of the magnetic field, B, varies strongly within plasma as a result of the small externally applied toroidal field.

The small external toroidal magnetic field yields a low value of the magnetic winding parameter, q, compared with configurations that employ a larger external toroidal field, such as the tokamak, ST, and stellarator. This parameter measures the helical twist of the magnetic field lines in the plasma. Thus, the RFP also offers unique science opportunities for plasma stability and confinement that complements the parameter regime of high-q configurations.

Scientific Contributions

Achieving the ITER-era goal will contribute significantly to fusion energy sciences and broader scientific disciplines. Examples are: (1) understand transport processes from turbulence that is primarily electrostatic, in the case where the ion gyroradius is relatively large, and the direction of the magnetic field lines changes rapidly within the plasma, (2) control of multiple instabilities in a plasma surrounded by a metal vessel of finite electrical conductivity, (3) understanding high-pressure plasmas exceeding stability limits locally or globally, (4) dynamics of magnetic self-organization in which magnetic reconnection, dynamo, momentum transport, and ion heating occur simultaneously, and (5) the generic linkage of these phenomena to plasma astrophysics.

Summary of Existing RFP Experimental Facilities And Other Available Means

There are four RFP experiments operating in the world. (1) The MST facility at the University of Wisconsin-Madison is the centerpiece of the US proof-of-principle RFP program. It is physically large in the RFP context (toroidal minor radius, a=0.5 m, and major radius, R=1.5 m), but has medium plasma current capability (I_p ~ 0.5 MA). The MST program emphasizes the study of plasmas with high confinement at high pressure made possible by control of the radial profile of the plasma current density and by auxiliary plasma heating. It also includes investigation of current sustainment by Oscillating Field Current Drive (OFCD), a low frequency form of magnetic induction. (2) The world's largest and highest power RFP facility is RFX-mod in Italy (a=0.46 m, R=2.0 m), with an achieved current of 1.5 MA and a design current of 2.0 MA. The RFX program emphasizes RFP performance at high current and active feedback control of resistive-wall instabilities with the most sophisticated sensing and feedback system in the world fusion program. A primary research goal is also optimization of magnetic self-organization to improve fusion performance. (3) A second, smaller European RFP experiment is Extrap-T2R in Sweden (a=0.18 m, R=1.24 m). It also has a sophisticated sensing and feedback system, and this program is focused mainly on advanced control techniques using this system. (4) A new, small aspect ratio RFP has recently commenced operation in Japan. The aspect ratio is the major radius divided by the minor radius. The RELAX device (a=0.25 m, R=0.51 m) aims to understand possible benefits of the low aspect ratio RFP, including the possibility for substantial pressure-driven "bootstrap" current.

A number of computational tools are available for RFP research. Owing to the importance of magnetohydrodynamics in RFP physics, state-of-the-art nonlinear resistive MHD computation has long been employed, e.g., the DEBS code. Nonlinear two-fluid (electron and ion fluids) studies are beginning to use new tools such as NIMROD. Gyrokinetic calculations have recently been initiated, adapting codes developed for tokamak research. Similarly, simulation tools developed for astrophysics studies are being adapted for RFP applications. The last major fusion power system study (TITAN) was completed around 1990. The TITAN codes are still available, but they are outdated and difficult to use.

Scientific Issues and Research Requirements

Eight scientific and technical issues for the RFP were identified by TAP. The research requirements for each of these are described below. Table 2 at the end of this section summarizes the issues and connections to theory and modeling, and facility capabilities.

TRANSPORT MECHANISMS AND CONFINEMENT SCALING

Key dimensionless parameters for transport mechanisms in the RFP are the Lundquist number, $S = \tau_R/\tau_A \sim I_p T_e^{3/2}/n_i^{1/2}$, and normalized ion gyroradius, $\rho^* = \rho_i/a$. In these expressions, τ_R is the magnetic diffusion time, τ_A is the transit time for Alfvén waves over the minor radius, T_e is the electron temperature, n_i is the ion density, and ρ_i is ion gyroradius, the size of the orbits of ions about magnetic field lines. In the limit where stochastic magnetic transport is dominant, the scaling of magnetic turbulence with *S* is crucial. In the limit where stochastic magnetic transport is minimized, electrostatic transport may be most important. This second limit has begun to be accessed using current profile control, which reduces the current-gradient drive for tearing modes, but such a limit may also be reached spontaneously at high *S*. If electrostatic transport is dominant, then as in configurations with a larger winding number, the main controlling parameters will be ρ^* and the collisionality, the rate at which particles collide with other particles.

The spontaneous reduction in magnetic turbulence resulting from a natural transition from a multi-helicity state (multiple tearing modes) to a single-helicity state (single dominant mode) is an emerging theme with potentially large impact on RFP confinement. Operation at higher current (and higher *S*) in RFX-mod indicates a natural preference (self-organization) for quasi-single-helicity conditions. Application of 3-D shaping or modifications of the plasma surface via external control coils might cause these states to be even more robust. The control of resistive wall modes in next-step RFP experiments requires a flexible set of active control coils, and these coils might also serve to control single-helicity transitions using 3-D shaping.

Presently MST is capable of $I_p \sim 0.5$ MA, with maximum *S* of about 10^7 (without current profile control) and $\rho^* = 0.013$, both quantities measured at the plasma center. RFX-mod is presently capable of $I_p \geq 1.5$ MA, with *S* of about 4×10^7 and $\rho^* = 0.007$. A burning RFP plasma, with an estimated $I_p \sim 20$ MA and $T_e = 10$ keV, will have much larger *S* of about 6×10^9 and much smaller $\rho^* \sim 0.002$. There also exists a large gap in computation and theory. Single-fluid computation has been used to study RFP dynamics with $S \leq 10^6$. At substantially higher *S*, two-fluid (electron and ion) physics is expected to be more important. Gyrokinetic calculations are only now beginning to be applied to the RFP. The capability for significant 3-D shaping will be important to understand single-helicity physics, both theoretically and experimentally.

Research Requirements

Understanding transport mechanisms and confinement scaling in the RFP requires a substantial extension in the Lundquist number and normalized ion gyroradius, both experimentally and in theoretical modeling. An upgrade to MST's power supplies may allow for modestly higher I_p (e.g., 0.8 MA), with *S* up to 4×10^7 and ρ^* down to 0.009. An upgrade already underway at RFX-mod will allow I_p of at least 2.2 MA, with *S* up to 3×10^8 and ρ^* down to 0.005. But to achieve the ITER-era goal, at least one RFP facility will be needed with capabilities well beyond that of MST and RFX-mod. One possible embodiment of such a facility in the US is envisioned as a two-stage experiment, ultimately capable of $I_p \geq 4$ MA, $S \sim 10^9$, and $\rho^* \leq 0.007$. This single facility could first be operated as an advanced proof-of-principle experiment with initially lower I_p and lower *S*. But with a substantial upgrade to its power supplies, it would then transition to a full-fledged performance extension facility to establish the basis for an RFP burning plasma in the ITER era. Basing two next steps on a single facility can hasten progress in RFP research and may prove less costly.

The critical physics that governs robust transition to a steady-state single-helicity state is not yet known, although high plasma current and good control of the magnetic boundary are observed to be important in present experiments. The possibility for 3-D shaping to induce robust single-helicity states needs to be explored theoretically and experimentally.

To achieve theoretical understanding and guide experiments, single-fluid computation should also be extended to higher *S*, ultimately including two-fluid physics. Gyrokinetic computation for the RFP will also be critical to predicting and understanding residual electrostatic transport in plasmas with reduced magnetic turbulence. Theory and modeling should span significant 3-D shaping capability, via spontaneous single-helicity transitions or externally applied magnetic fields.

CURRENT SUSTAINMENT

The steady-state sustainment of the plasma current in the RFP is challenging. Unlike the tokamak, the spontaneous plasma-pressure-driven "bootstrap" current in the RFP tends to be small, even at sufficiently large pressure for a high-power-density reactor core. While an efficient magnetic induction scenario using long pulses of plasma current separated by short temporal gaps to reset the transformer is plausible, deriving advantages from the RFP's low external field, steady-state would be preferred. A novel form of steady-state operation called OFCD is under investigation. For OFCD, very low (audio) frequency AC inductive voltages are applied, and a DC current is created and sustained through magnetic self-organization processes. While this is still magnetic induction, the applied voltages are AC, and the magnetic flux in the transformer does not accumulate in time. To date, 10% current drive by OFCD has been experimentally demonstrated, and nonlinear resistive MHD computation indicates 100% current drive is possible. MST has power supplies specially built to provide the large AC voltages required for OFCD experiments, but MST's relatively low *S* and relatively short pulse duration will likely limit the current fraction driven by OFCD to ~ 20%. The existing power supplies for RFX-mod have less capability than MST's for AC voltage production.

Substantial bootstrap current might be possible in the RFP if the pressure can be increased substantially beyond presently achieved values. The pressure limit (or beta limit, normalizing to the pressure associated with the confining magnetic field) in the RFP is not yet identified, in part due to a lack of sufficient plasma heating power in present experiments. As with tokamak plasmas, the bootstrap current would likely be maximized at low aspect ratio, a barely explored parameter regime in RFP research.

Research Requirements

A definitive experimental demonstration of OFCD requires a plasma with a sufficiently high Lundquist number and long pulse duration. A primary concern is that the induced AC oscillation of the confining magnetic field must be small enough to avoid unacceptably large dynamic variation of the plasma equilibrium during the course of an OFCD cycle. Experiments with a Lundquist number $S=10^{7\text{-}8}$ are projected to be required to maintain the oscillation in the plasma equilibrium within the range of established RFP operation, to avoid effects such as additional tearing mode instabilities that would not normally be present. It is also necessary that the plasma (and OFCD) duration be at least comparable to the current relaxation time scale, so that the full effect of the current drive is established. Note that the current relaxation time also increases with the Lundquist number.

Upgrades to the existing RFP facilities would be useful to demonstrate larger-fraction current drive by OFCD, in particular agile power supplies capable of producing a wide range of inductive voltage waveforms. This would permit assessment of critical physics issues and validation of theoretical models with gradual investment. However, an advanced proof-of-principle facility that provides access to high plasma current, high Lundquist number, and long pulse is necessary to demonstrate full OFCD sustainment. Control of the plasma-material boundary interface beyond that available in present facilities might be crucial to maintaining high-quality plasmas with low impurity content.

Nonlinear resistive MHD and two-fluid computation is a major research tool needed to understand OFCD. The magnetic self-organization process that OFCD requires is anticipated to involve MHD tearing instabilities. Furthermore, the very low frequency of the AC voltages implies that computation must span a large range of disparate temporal scales. These are demanding needs, requiring state-of-the-art computational tools that include relevant physics with low dissipation ($S=10^{7-8}$). At high temperature, it is also anticipated that two-fluid physics will become important. Exploration of high beta at low aspect ratio is needed. While a specialized concept exploration RFP experiment might eventually be required, collaboration on existing smaller low-aspect-ratio devices such as RELAX or LTX should be considered. Present RFP experiments with volume average beta of $\langle\beta\rangle$ =12% exceed some theoretical limits for pressure-driven instability, indicating a need for improved understanding.

INTEGRATION OF CURRENT SUSTAINMENT AND GOOD CONFINEMENT

Current sustainment in the RFP could be (i) steady-state using OFCD, (ii) quasi-steady-state using a hybrid combination of OFCD and inductive current profile control, or (iii) purely pulsed. Achieving good confinement in at least one of these cases is essential, building on the resolution to the confinement and current sustainment issues described above. Noninductive current drive methods using radiofrequency or neutral beam injection might be effective for targeted current profile control, if power requirements are not excessive.

Research Requirements

The demonstration of 100% current sustainment with OFCD and addressing the scaling of confinement with OFCD is unlikely to be accomplished in present relatively low-*S* experiments. Numerical simulations presently do not reach the Lundquist numbers specified for the RFP ITER-era goal, nor do they employ the necessary two-fluid physics. Equally important, present simulations do not incorporate transport calculations. This is needed not only to predict transport with OFCD, but also to understand the effect of varying transport (and plasma temperature and electrical resistance) on the effectiveness of OFCD. Gyrokinetic simulations are not yet available to study the compatibility of electrostatic transport and current sustainment.

At least one new facility is needed with capabilities well beyond those of MST and RFX-mod. This facility would provide higher *S* as well as longer pulse duration. At least initially, the plasma current would likely be sustained primarily by conventional induction, followed by substantial upgrade(s).

PLASMA BOUNDARY INTERACTIONS

The existing RFP facilities are not well equipped to investigate advanced control of the plasma boundary. Such control is important to prevent damage to the device wall, and to prevent influx of impurities into the plasma. Although some knowledge is transferable from research in other concepts, there are challenges unique to the RFP. Existing RFP devices employ basic material limiter structures to protect the wall from plasma interaction, and particle control is maintained by fueling adjustments and limiter conditioning. Existing devices are also not easily modified to investigate magnetic divertors, which allow heat flux from the edge plasma to be spread over a large area, rather than be concentrated on a limiter. Components for other strategies like pumped limiters or a liquid wall might be testable in present devices. This is a critical area of need, and the compatibility of boundary control solutions with confinement, sustainment, and global stability challenges must be determined. Significant boundary control capability should be considered for any new device.

Research Requirements

New devices or major upgrades will be required to adequately address the plasma-boundary issue. This research could begin in smaller devices at the concept exploration level. Possibilities for investigation are a toroidal magnetic divertor or liquid lithium boundary. The effect of plasma cross-sectional shape could be simultaneously investigated. Since ohmic heating for smaller RFP plasmas is substantial, it is relatively simple to efficiently attain heat flux levels relevant to future devices in small concept exploration experiments.

Some progress can be made in existing RFP facilities by improving diagnosis of the edge and core with the goal of increasing understanding of the 3-D nature of the edge plasma, impurity influx and production, and impurity transport. This will require a variety of probe and spectroscopic instrumentation, and may use active techniques such as impurity injection. Going forward, a key component of the effort to understand and control the plasma boundary will be modeling of the edge plasma, boundary interactions, and impurity transport. This modeling effort should be coordinated among present RFP experiments and should take advantage of the substantial tools and understanding available in the larger fusion community. It should also be integrated with the broader scope modeling and simulation effort that will be required to understand RFP physics.

ENERGETIC PARTICLE EFFECTS

The confinement of energetic particles (especially ions) and their affect on stability are largely unexplored for the RFP. Although some knowledge, both experimental and theoretical, is transferable from other configurations, unique physics is also expected. While particle orbits are more nearly classical in poloidal-field-dominated plasmas like the RFP, concerns include direct orbit losses at lower magnetic field and nonclassical loss associated with magnetic turbulence. The planned installation of ~1 MW neutral beams in MST and RFX-mod will provide first-time assessment of energetic particle effects in the RFP with the ion speed comparable to Alfvén speed, and with the energy content comparable to that of the plasmas.

Research Requirements

Installation of neutral beams at larger energies, currents, and time duration are required to study the effects of significant populations of super-Alfvénic particles. The energy content and momentum of the beams must be comparable to or higher than that of the background plasma. The beam duration must be longer than both the plasma confinement time and the fast ion confinement time. Theoretical and numerical tools, such as TRANSP, NOVA, and ORBIT, require adaptation for the RFP configuration.

DETERMINING BETA LIMITING MECHANISMS

The required value ~20% of poloidal beta (ratio of plasma pressure to poloidal magnetic field pressure) for an attractive RFP reactor has already been experimentally achieved, but complete theoretical understanding is lacking. It is therefore not yet possible to project beta limits in a burning plasma. Optimization and control of the plasma pressure profile, possibly as a function of plasma cross section and aspect ratio, may become important to avoid instability and maintain the high beta necessary to achieve the ITER-era goals.

Research Requirements

By upgrading the existing facilities, especially the neutral beam injection system in terms of power and fueling rate, it is possible that the beta limit could be identified at moderate S for both axisymmetric and helical RFP equilibrium. However, the plasma parameters (energy confinement time, fast ion slowing down time, and plasma duration) make fast ion thermalization difficult, most likely limited to higher density operation provided by pellet injection. Dependence of beta limits on cross section and aspect ratio cannot be studied on the existing facilities, requiring new facilities to explore geometric optimization.

Testing beta limits at the larger S available in an advanced, long-pulse proof-of-principle facility and a subsequent performance extension facility with full heating and fueling capabilities is essential in achieving the RFP goal.

Comprehensive theoretical analyses are required beyond the standard Suydam criteria (critical pressure gradient), which have been surpassed by experimentally achieved values. Extended capability for computational tools at larger S and with two-fluid physics is also required, in conjunction with theory, to confirm and quantitatively predict the beta limits.

ACTIVE CONTROL OF MHD INSTABILITIES

Both resonant and nonresonant MHD instabilities play an important role in the RFP. The control of these instabilities and their consequences by means of non-axisymmetric coils (both saddle and flux loops) is an important area of investigation, since this may be essential in any major next-step RFP device and a future reactor. The T2R and RFX-mod facilities in Europe have significant active control experimental capability used to completely stabilize resistive wall modes. This establishes the critical physics requirement that RFP plasmas be maintained beyond the time scale of passive stabilization by an electrically conducting wall. In addition, saddle loops located at cuts or breaks in the conducting vessels surrounding the plasma have been used to mitigate localized field errors that can cause wall locking of tearing modes, as shown in RFX-mod and MST. RFX-mod experiments have also established that locking of tearing mode phases to the wall can be mitigated by appropriate feedback using the full-coverage coil set.

Research Requirements

Present experiments suggest that it is possible to operate a RFP without a very conducting wall. However, key questions remain, and the optimal control system has not been identified. Will coils located inside the vessel be able to mitigate phase locking to the wall, optimize the plasma shape and help for power handling (as in ITER)? Will it be necessary to include special flux loops to control m=0 modes (m is poloidal mode number)? Will ad hoc saddle coils be able to compensate for the residual error fields due to the unavoidable asymmetries of the magnetic boundary? Certain types of coils may be optimal for different instabilities. Nonlinear coupling between m=1 and m=0 poloidal mode makes control of both harmonics important. Separate coils optimized for m=0 control may be needed. The toroidal field coil set in RFX-mod could in principle provide significant m=0 mode control, but the precision of the currently installed power supply units is limited. In both RFX-mode and Extrap T2R, the issue of sideband aliasing (unwanted perturbations in other modes) has been observed and improvements in control have been obtained by reducing it either by increasing the number of actuators (T2R) or by removing it from the measurements in real time (RFX-mod). It is vital to understand the required minimum number of control coils, and how advanced control algorithms can surmount issues such as sideband aliasing.

The development of accurate models is essential for understanding the behavior of the present control systems and for predicting the performance and control requirements of future devices. Several approaches are possible. An existing model of the electromagnetic boundary that includes simplified transfer functions for the feedback-controlled current power supply units in a multi-input, multi-output paradigm reveals the necessity to better understand and model the mutual couplings of the feedback coil hardware. In the near term, merging the electromagnetic code Cariddi with the MARS code can advance this issue. The development of a full simulator of an RWM control system for existing devices will also allow testing different kinds of model-based control approaches, where standard PID controllers will be replaced by more complex state-space controllers, and which may also take into account the 3-D structure of the magnetic boundary. A parallel approach to RWM control modeling is based on the development of a more complete physics model of the plasma in a simplified geometry, taking into account plasma pressure, compressibility, inertial, dissipation and longitudinal plasma rotation. In addition to RWM modes, optimization of the magnetic boundary for the control of tearing modes is important, for example, identifying the ideal non-axisymmetric deformation of the last closed magnetic surface.

SELF-CONSISTENT REACTOR SCENARIOS

In addition to the integration of current sustainment and improved confinement, a self-consistent scenario must also integrate RWM control and plasma boundary control to achieve high-performance RFP discharges and establish the knowledge base for a burning plasma experiment. The RFX-mod and Extrap T2R facilities have capability to begin addressing the optimization of RWM control, but not for issues such as requirements on the proximity of feedback coils to the plasma surface. MST can apply current profile control and possibly enhance plasma pressure, but cannot include reactor-relevant resistive wall mode stabilization (MST operates with a thick conducting wall), and is limited in OFCD capability. In general, the compatibility of boundary control is a large issue that cannot be addressed in present facilities.

Research Requirements

Experimental capabilities. The development of self-consistent scenarios requires a facility that includes all the aforementioned capabilities, probably at the performance extension level. This includes the capabilities for 100% current sustainment, heating, fueling, and RWM and edge control. Testing feedback coil proximity and plasma rotation might be possible through extensive modification of existing facilities, or perhaps with a new, smaller facility optimized for this purpose.

Reactor study capabilities. Further system code study is essential to develop such self-consistent scenarios, including both steady-state solutions and hybrid solutions with an OFCD phase and a current ramp-down phase. Assessing the impact of special constraints imposed by the reactor environment is important, e.g., the proximity of active feedback coils if they need to be located outside the neutron shield.

Numerical capabilities. Three-dimensional nonlinear fluid computations, which could incorporate a toroidal divertor, are required to develop, predict and optimize self-consistent reactor scenarios.

Issues	Existing Facilities		Needed Facilities			Existing Theory, Computation, Modeling	Needed Theory, Computation, Modeling
	MST	RFX	Advanced Control PoP	CEs	PE (ITER-era Goal)		
Identify transport mechanisms and establish confinement scaling	current profile control, *S*-scaling at I_p<0.5 MA	*S*-scaling, single-helicity state, at I_p <2 MA	current profile control, *S*-scaling, single-helicity state, at I_p≥2 MA	single-helicity state at low aspect	establish base for burning-plasma experiment with I_p≥4 MA	3-D nonlinear MHD, test-particle stochastic theory	3-D nonlinear 2-fluid ($S=10^8$), gyrokinetics
Current sustainment	partial OFCD, pulsed	pulsed	OFCD, pulsed, and hybrid-inductive	optimize bootstrap current at low aspect ratio	OFCD, pulsed, or hybrid-inductive	3-D nonlinear MHD ($S=10^6$)	3-D nonlinear 2-fluid ($S=10^8$)
Integration of current sustainment and improved confinement	extend current profile control duration (inductive, rf)	*S*-scaling, single-helicity state (pulsed)	long-pulse profile control, *S*-scaling for OFCD/pulsed	limited capability	integrated high performance, long pulse	combine above	combine above
Plasma boundary interactions	limiter	limiter	begin with limiter, explore divertor and other concepts at a later stage	test options: toroidal divertor, pump limiter, liquid wall, etc.	integrated control	relevant capability from larger fusion community	adapt for RFP configuration
Energetic particle effects	tangential beam injection, substantial fast ion beta, $v_i/v_A≥1$	perpendicular beam injection, $v_i/v_A≤1$	beam injection with $v_i/V_A >> 1$	limited capability	fully assess energetic particle effects	Relevant codes, e.g., NOVA, ORBIT, etc., exist	adapt for RFP configuration
Determine beta-limiting mechanisms	possibly identify at I_p<0.5 MA	possible with further improved confinement	identify at I_p≥2 MA	limited capability	verify and control at high performance	3-D nonlinear MHD with finite pressure	3-D nonlinear 2-fluid
Active control of MHD instabilities	limited capability	significant capabilities for m=1 modes, while limited for other modes	optimized capabilities	optimized capabilities	integrated control	linear and 3-D nonlinear MHD	none known
Self-consistent reactor scenarios	limited capability	limited capability	limited capability	limited capability	substantial capabilities	TITAN system study codes	3-D nonlinear model in realistic geometry (e.g., toroidal divertor)

Table 2

THE COMPACT TORUS: FRC AND SPHEROMAK

Introduction

The mission of Compact Torus (CT) research is to:

Develop a compact magnetic fusion reactor without toroidal field coils or a central solenoid.

Progress in this mission is supported by the ITER-era goal established by TAP:

To demonstrate that a CT with a simply connected vessel can achieve stable, sustained or long pulsed plasmas at kilovolt temperatures, with favorable confinement scaling to proceed to a pre-burning CT plasma experiment.

A CT has a toroidal magnetic and plasma geometry, but is contained within a simply connected vacuum vessel such as a cylinder or sphere. Spheromaks and field-reversed configurations fall into this category. Compact tori have high plasma and engineering betas, potentially obtaining high-power density at relatively low magnetic fields. The primary benefit of CTs for fusion is the absence of toroidal field coils, simplifying solutions to many engineering problems. Studying high-beta, fusion-relevant plasma in simply connected geometries affords both physics and technology opportunities not found in other configurations, in a program in which the US is the world leader.

The CT program currently consists of experiments at the concept exploration level, involving both the spheromak and the FRC, and other hybrid configurations may be possible. CT funding has traditionally been a small part of the alternate concepts program, but both the spheromak and FRC configurations have made substantial plasma physics progress in recent years, pointing the way to future research on CT science and technology opportunities.

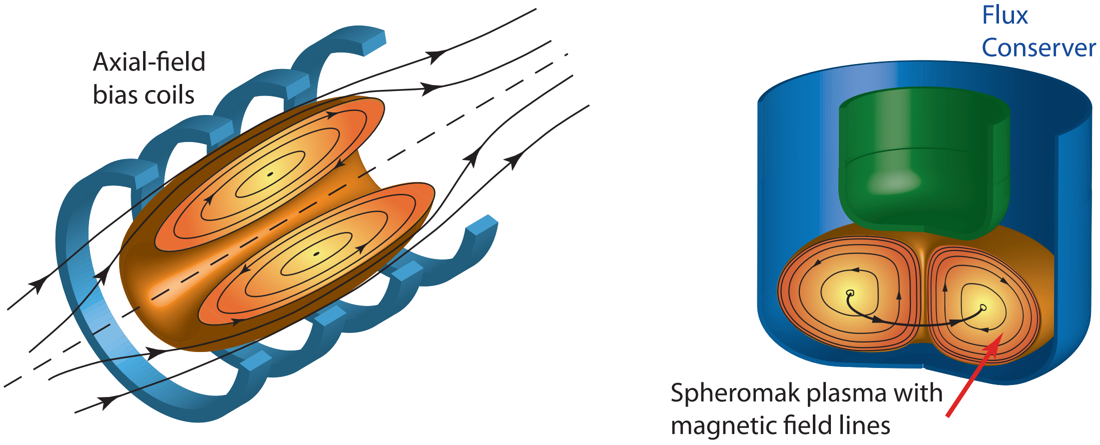

Figure 6. On the left is the field-reversed configuration and on the right is the spheromak configuration. No physical structure links the plasma torus in these configurations.

Scientific Issues and Research Requirements

Three primary challenges need to be addressed in the ITER era: (1) formation and stability in the fusion-plasma regime, (2) transport and energy confinement, and (3) efficient current drive and flux sustainment. The FRC and spheromak operate in significantly different physics regimes, so the Issues and Research Requirements for the two configurations are discussed separately. Additional sections consider diagnostics, theory and modeling, and reactor studies for CTs. Tables 3 and 4 at the end of the section summarize the needed facilities and the theory and codes to handle the research requirements discussed below.

Connections to Broader Science

The CTs are closely coupled to a wide range of broader science. They extend fusion plasma science to very low aspect ratio and minimal external magnetic field, with no toroidal field coils. The FRC has no toroidal field in the ideal limit, and the magnetic winding ("safety") factor of the spheromak lies between the tokamak and RFP. A magnetic dynamo drives the spheromak toward a self-organized state in a complementary way to the RFP; in both, this science is closely related to processes in space and astrophysics. Much of the research is done in universities and the configurations lend themselves to the "discovery science," which may provide transformational ideas in the search for fusion energy.

FRC

STABILITY AT LARGE *S*: ACHIEVE GLOBAL STABILITY AT LARGE *S* IN LOW COLLISIONALITY FRCs. *(S IS THE NUMBER OF ION LARMOR RADII BETWEEN THE CENTRAL FIELD NULL AND THE SEPARATRIX)*

The highest priority issue for the FRC is maintaining stability as the size is scaled up and the plasma becomes more MHD-like. In the largest theta-pinch FRC, the Large s Experiment (LSX), hot, high-density stable FRCs were formed with s up to 4, but s ~ 20 or more will likely be needed for reactors. Cold plasmas in SSX had s ~ 10 and were unstable at slower than MHD growth rates. The most worrisome FRC instability, the n=1 axial tilt, is external in oblate FRCs and can be stabilized by close-fitting conducting walls or strong external field shaping. In prolate FRCs the MHD-tilt is internal, and other stabilizing means are needed. Experiments and simulations to address this issue can be handled in part by existing facilities and codes, but their capability falls short of the requirements to resolve it. The limitations of existing simulations and experiments constitute a major research gap.

Research Requirements

New facilities and/or significant upgrades of existing facilities and extended or new codes will be needed. The parameters of the required experimental facilities include:

- Low-collisionality FRC plasmas with sufficient flux to confine fast ions, ~ 20-50 mWb.
- Efficient heating and current drive methods.
- Fast ion creation at mid to lower 10^{19} m^{-3} densities, possibly by neutral beam injection, to generate large fast ion-ring currents for stabilizing effects and for current drive if needed.

- A sufficiently high confining magnetic field to reduce the ratio of electron drive velocity to ion thermal speed, $\gamma_d = v_{de}/v_{ti}$, below unity.

It can be shown that *s* ~ $RB/T^{1/2}$ ~ $Rn^{1/2}$ in a high beta FRC ($\beta=2\mu_o n[T_e+T_i]/B_e^2\approx 1$), where R is the plasma radius, B the magnetic field and B_e the field outside the plasma, T the average temperature, and n the average density. RB is thus a critical engineering parameter and needs to be increased by a factor of several from recent experiments to study high-*s* stability. To keep sustainment powers reasonable, densities should be in the low 10^{19} m^{-3} regime. To achieve high *s* values in steady-state FRCs, the magnetic-field separatrix radius should be of order 0.5-1 m; a poloidal flux ~ 20-50 mWb will be needed to confine energetic (10-20 keV) ions from NBI to study their stabilizing effects. Improved diagnostic capabilities, especially for density, temperature and magnetic field profiles, and for *s*, are urgently needed to resolve internal physics.

There are three ways of forming such FRCs, and one or more can be used in experiments to close this gap. (1) FRC formation employing a field reversed theta pinch, together with translation, has been successful in forming hot stable FRCs at densities appropriate for neutral injection. Confinement scaling for these FRCs indicates that an LSX-scale translation experiment could provide both the flux and confinement required for beam trapping and sustainment experiments. (2) Merging pairs of 5 mWb spheromaks have formed FRCs with over 5 mWb of flux. Much larger (up to 75 mWb) spheromaks have been formed, suggesting that it is possible to employ this method to create large flux FRCs. Transformer drive could be added to sustain flux in a physics experiment. (3) The scaling of rotating magnetic field current drive (RMF) to form and sustain FRCs is now reasonably well understood, yielding FRCs with diameters and fluxes ranging from ~5 cm and ~10 μWb to ~ 75 cm and 5 mWb. Scaling up another factor of 2.5 in diameter would be relatively straightforward, although new issues may arise as the plasma size and density increase. If the resistivity scaling with γ_d continues as predicted, higher fluxes and temperatures might be sustained with MW-level sustainment powers similar to those used in present experiments.

Besides tangentially-injected NBI, it would be useful to have the capability to study the physics of toroidal flow shear, perhaps through a combination of RMF and NBI, which produce momentum in opposite directions; and to study both oblate and prolate FRCs, including passive stabilizers needed for the oblate FRCs.

An expanded simulation effort closely coupled to the experiment is needed to guide, understand and optimize experiments. Simulations need to be three-dimensional; for example, the RMF-driven FRC is 3-D in nature. There is a need to develop 3-D codes, e.g., hybrid (MHD+kinetic), kinetic, PIC, and MHD, to explain and predict experimental results. Such code development is already underway at a low manpower level, showing promising agreement with experiments.

TRANSPORT: REDUCE TRANSPORT IN KEV FRC PLASMAS AND UNDERSTAND TRANSPORT SCALING.

Present experiments observe anomalous resistivity and associated anomalous transport. There is no present capability to study transport at high flux and large radius where the anomalous transport is anticipated to be reduced. This lack of capability is an important gap.

Research Requirements

New facilities and/or significant upgrades of existing facilities, and extended or new codes will be needed.

Theoretical and experimental evidence that the anomalous cross-field resistivity is related to γ_d supports the hypothesis that resistivity is due to lower hybrid drift waves or a related mode. A drift-generated resistivity from lower hybrid drift waves has been calculated theoretically and found to explain simple theta-pinch plasma profiles. A hot, steady-state high-*s* experiment will have $\gamma_d < 1$, may address the transport issue experimentally, and could provide a platform to study confinement scaling. As auxiliary heating methods may alter transport, several such techniques should be studied. Effects of fast ions on transport should also be addressed. Experiments need to be supported by theory and simulation.

Plasma fluctuations associated with the diamagnetic current drift must be studied with noninvasive diagnostics. Global confinement time studies require multi-point diagnostic systems. Exploration of this issue may be undertaken by operating the current density low enough in the high-*s* experiment that the modes thought to be responsible are not excited. Additional diagnostics may be required, and additional experimental facilities may advance these studies, particularly the effects of plasma shape and flow shear. If the resistivity is found to approach classical, new energy and particle loss mechanisms may become dominant; it is uncertain whether the high-*s* facility will continue to be appropriate or a new experiment will be needed.

CURRENT DRIVE AND SUSTAINMENT: ACHIEVE AND UNDERSTAND EFFICIENT CURRENT DRIVE AND SUSTAINMENT OF KEV FRCs WITH GOOD CONFINEMENT

FRC flux sustainment requires overcoming the ohmic losses associated with maintaining the diamagnetic toroidal currents. However, the cross-field resistivity, $\eta_{\perp}$, is found to be anomalously high and needs to be decreased for sustainment at low power. Existing experimental facilities are capable of studying some aspects of RMF and merging spheromaks, although the latter cannot presently study steady-state current drive, and are limited in capability. Unless the experiment used to study large-*s* is capable of long-pulse current drive and sustainment, a new experiment is needed to extend present work and fill this gap.

Research Requirements

FRC toroidal current is essentially diamagnetic, so it should be possible to sustain FRC poloidal flux by supplying enough particles deep in the core, perhaps by neutral beams. Some other form of current drive will still be required on axis. RMF could provide the current and neutral beams might provide both current and particles.

Steady-state current drive for longer than 10 ms has been demonstrated using RMF in FRCs. Fast ions from tangentially aimed neutral beam injection were used in field-reversed mirror experiments, resulting in significant reduction of the magnetic field and peak $\beta > 2$ but not field reversal; however, injection into a reversed field is expected to sustain the configuration. Energetic ions also stabilize MHD modes, due partly to FLR effects but primarily as the wave-beam ion response can be opposite in phase to the thermal plasma response, thereby reducing the instability growth rate. Simplified stabilization conditions are that the average toroidal rotation frequency of the beam ions be larger than their axial betatron frequency and the fast ion current be sufficient to balance the instability drive.

Code development for RMF should be extended to include wave physics that may become important as the plasma dimensions increase and should be coupled to other codes to obtain a more consistent calculation, including as much physics as possible.

FAST PARTICLES: UNDERSTAND EFFECTS OF FAST PARTICLES ON CURRENT DRIVE, STABILITY AND CONFINEMENT.

Fast particles, e.g., due to neutral beam injection, are needed to stabilize the FRC in the MHD regime, but are likely to have other effects on the plasma. A start on this issue will be made during study of higher priority issues.

Research Requirements

There are significant effects that will require additional study by theory and experiment, including the coupling of the energetic ions to plasma magnetosonic and other modes, plasma potential due to first orbit losses and plasma rotation, and effects of non-Maxwellian distributions on stability.

Facilities capable of research on current drive, sustainment, transport and stability are expected to be capable of addressing the effects of fast particles. Appropriate diagnostics will be needed.

HEATING: DEMONSTRATE EFFICIENT HEATING METHODS (E.G., ROTATING MAGNETIC FIELDS, NEUTRAL BEAM INJECTION, AND COMPRESSION) TO INCREASE THE TEMPERATURE TO NEAR BURNING-PLASMA VALUES

Experiments to date have focused on formation and sustainment, and not on heating the plasma, but ion heating will be required to reach fusion temperatures.

Research Requirements

Experiments and simulations that research heating by these methods are needed. A facility capable of current drive, transport, stability and sustainment research should also be able to address plasma heating in FRCs, although upgrades may be required.

SPHEROMAK

SUSTAINMENT: DEVELOP AND DEMONSTRATE METHODS FOR SUSTAINING A SPHEROMAK WHILE ACHIEVING GOOD ENERGY CONFINEMENT

Several formation and steady-state sustainment techniques have been and are being used, including injection of helicity (linked toroidal and poloidal magnetic fluxes) from an electrostatic gun; helicity injection from crossed, inductively driven injectors; and induction using magnetic coils. However, the efficiencies of current drive are too low for steady-state spheromak experiments.

Small, concept exploration experiments and resistive MHD simulations are testing innovative ideas. Exploratory research is important and should continue, but cannot test ideas in a fusion-quality environment. Lack of such an experiment and supporting theory is a major gap in the spheromak program.

Research Requirements

Advancing the spheromak program requires an experiment with the high-quality vacuum environment, long-pulse capability, flexible power systems, comprehensive diagnostics, and closely coupled theory needed for high-quality research in fusion plasmas. It is a high priority to determine the detailed features of this experiment and to construct it to address the issue of sustainment and other important spheromak research.

Magnetic relaxation has sustained spheromaks, and spheromak research has contributed significantly to self-organized plasma research. However, the accompanying magnetic fluctuations have an impact on confinement, as shown in laboratory and resistive MHD simulations. The rotating n=1 "column" mode (with nonlinear harmonics) that is excited by DC, coaxial helicity injection has been shown to be detrimental to closed magnetic flux surfaces. Scaling of fluctuations and confinement with plasma temperature and magnetic field strength is an active area of research, so the possibility of relaxation-based current drive remains open. Nonetheless, the ITER-era goal compels investigation of new sustainment methods that reduce or avoid relaxation over at least part of a discharge cycle.

Two approaches have been proposed for addressing this issue: (a) Developing better helicity injection or other steady-state, efficient current drive techniques that are compatible with good confinement, and (b) periodically rebuilding the plasma flux and current separated in time from a "coasting" phase, which experiments have already shown to have good confinement. Research needs for these are discussed below.

> (a) Plasma currents in the spheromak can be generated (and sustained) by a variety of schemes. Important issues are minimal perturbation of confinement, efficient transfer of electrical energy to stored plasma energy, and favorable scaling to the next step. Known spheromak attributes could be exploited in different ways: for example, different steady-state helicity-injection techniques, pulsing, merging spheromaks, use of solenoids, and auxiliary current drive using neutral beams or radiofrequency. It may be useful to harness higher order modes and have more control over reconnection, or to use auxiliary current profile control, heating or externally applied magnetic fields to influence the conductivity profile. Non-axisymmetric methods such as the inductively driven concept need further study to evaluate their promise fully. Schemes more commonly considered for other concepts, such as oscillating field current drive, could also be considered.
>
> (b) An alternate approach is to separate the plasma sustainment and confinement phases in time. This was demonstrated for coaxial helicity injection on SSPX where it was termed refluxing, and in principle could be applied to inductive helicity injection techniques. In refluxing, periodic current pulses build spheromak flux and are followed by slowly decaying flux and current with closed magnetic surfaces that confine plasma well. In a reactor,

this would be a burn phase, so an assessment of the reactor potential is desirable for this concept. Research needs include extending existing simulations to have as much physics as possible and then using the results of this research to design a new experiment. This experiment will need at least the capabilities described in (a); both approaches might be explored on the same facility. Achieving a maximum confinement-phase pulse length may require developing current-profile control techniques to prevent the formation of large magnetic islands at low-order resonant surfaces in the plasma.

Nonlinear macroscopic stability theory and computations with two-fluid and kinetic effects can have a large role in testing and understanding new approaches to sustainment. However, the computational challenges associated with modeling relaxation fluctuations increase with plasma performance (measured by dimensionless parameters such as the Lundquist number). While implicit methods help address stiffness, they do not alleviate the need for resolving multiple temporal and spatial scales. Increasing access to computing facilities that serve communication-intensive applications is needed. Integrated simulation with self-consistently modeled radiofrequency can be applied to profile control. The work of the Fusion Simulation Project can be applied if it is sufficiently general to include low-field configurations.

FORMATION: DEVELOP AN EFFICIENT FORMATION TECHNIQUE TO ACHIEVE FUSION-RELEVANT SPHEROMAK MAGNETIC FIELDS

The efficiency of spheromak formation needs to be improved. Small experiments are exploring formation techniques, but are not capable of demonstrating high efficiency due to vacuum and other limitations, and lack of an adequate experiment is a significant gap for formation research. Computational gaps for formation studies are similar to those for sustainment studies.

Research Requirements

Helicity injection has demonstrated the capability of generating MA spheromaks and achieving 75 mWb poloidal fluxes. However, efficiencies need to be increased significantly from the present 5%-20% as the magnetic energy is almost entirely supplied by plasma currents and needs to be increased substantially in future research. Furthermore, the amplification of the applied poloidal bias flux is less than ten in existing and past experiments, allowing a relatively large volume of open magnetic field lines on the edge where ohmic losses are high. Reactor estimates indicate that the flux amplification must be increased by almost an order of magnitude. Simulations indicate that if the bias flux and drive current are reduced together following formation, then the resulting plasma has low losses on edge magnetic field lines and reduced instability drive at resonances inside the separatrix; this and other innovative ideas should be explored further.

Regardless of the approach, the dissipation must be minimized to achieve an efficient ramp-up and therefore during ramp-up the configuration must confine heat reasonably well. Ramp-up schemes must be found that minimally affect confinement, so long as energy and helicity lifetimes are maintained during formation. Understanding the coupling efficiency of the external supply to the spheromak is critical to optimize buildup scaling.

TRANSPORT: DETERMINE UNDERLYING TRANSPORT MECHANISMS AND CONFINEMENT SCALING IN LOW COLLISIONALITY SPHEROMAK PLASMAS

Core electron energy confinement in quiescent spheromaks has been demonstrated to be comparable to tokamak L-mode values, but the mechanisms are not understood and there are no measurements of ion transport. There are presently no experiments capable of spheromak transport studies. This is an important research gap.

Research Requirements

New experiments having long pulse lengths and high magnetic flux are needed. An experiment at least the size of SSPX ($R_0 > 0.3$ m, $I_{plasma} > 500$ kA) will be needed so that charge exchange is not a dominant loss and so there is enough power to burn through impurities. Supporting theory and simulations are also important. Diagnostic capability is essential, e.g., to differentiate between transport driven by magnetic and electrostatic fluctuations. This research will broaden our scientific knowledge by extending confinement studies to the safety factor range $\sim 0.2 < q < 1$ lying between the RFP and the tokamak.

Energy confinement experiments in slowly decaying plasmas yield electron thermal conductivities in the L-mode range, although the database is limited. Power-balance calculations with 0.1 keV < $T_{e,peak}$ < 0.5 keV and densities ~ 10^{20} m^{-3} yield electron thermal conductivities ~1 m^2/s - 10 m^2/s in the core of the spheromak. Because the experiments were at the concept exploration level, there were no measurements of ion temperatures or ion thermal conductivity; future work will need ion confinement diagnostics. The electron transport mechanism is unknown, although it was found that the best confinement occurred on SSPX when the q-profile did not cross low-order resonances. Resistive MHD simulations support this result: in slowly decaying plasmas, closed magnetic surfaces form and the electron temperature rises to a few hundred eV. If the q-profile evolves to cross low-order surfaces, islands form; when they overlap they result in stochastic magnetic field lines and high thermal losses throughout much of the spheromak. On CTX the best confinement occurred soon after helicity injection was stopped and the equilibrium decayed slowly. The spheromaks ohmically heated to $\beta_{e,peak} > 20\%$ before a strong pressure-driven interchange occurred, demonstrating ohmic heating to the beta limit. In the limit of macroscopic (MHD-like) dynamics being eliminated, microturbulence may govern cross-field transport. How this occurs in high-β, low-q equilibria will be a new area of research.

Rotation and flow shear seem to have a very positive effect on the stability and confinement of tokamak and ST plasmas, and should be applied to a high-temperature spheromak, perhaps by using neutral beam injection. Flow shear may also be used to control turbulence caused by relaxation or drift modes.

BETA LIMITS: UNDERSTAND BETA-LIMITING MECHANISMS

Past experiments have demonstrated high peak and volume-averaged betas (5-20%), but the limiting mechanisms are not well understood. No present experimental facility is capable of addressing this issue.

Research Requirements

The experiment needed for sustainment and transport studies, or an upgrade thereof, may provide the required capability. In addition, significant auxiliary heating is needed to study beta limits.

Experiments in SSPX found a limiting electron beta, ~ 5% (peak), although the highest temperature shots exceeded this by up to a factor of 2. The ion contribution is not known, but could contribute up to a similar factor. It is also unknown whether the achieved betas were limited by ohmic power balance or by the onset of pressure-driven modes. The transient peak electron beta ~ 20% observed in CTX suggests that the SSPX results were limited by ohmic power. There are no comprehensive computational studies of beta limits, though nonlinear simulations that evolve pressure are consistent with experiments. Understanding this limit and the effects of spheromak shaping may allow it to be increased significantly. Such a result would significantly improve the attractiveness of a spheromak reactor. Both simulations and experiments are needed to address this issue.

PARTICLE BALANCE AND DENSITY CONTROL: UNDERSTAND PARTICLE BALANCE AND CONTROL OF PLASMA DENSITY AND IMPURITIES

Particle balance and density control has not been systematically studied in spheromaks, and no present experimental facility is capable of addressing the associated science.

Research Requirements

Particle balance and density control may require a specialized facility or significant upgrades (including diagnostics) to the experiment described above. Achievement of the ITER-era goals will likely require extension of techniques necessary for previous spheromak experiments.

Particle and density control is a complex scientific issue, involving plasma-wall interactions, penetration of the plasma by neutral particles, density pinching, plasma flows, etc. The minimum spheromak operating density may be limited to a constant fraction of the current density, perhaps due to stability related to electron streaming. Experiments with wall conditioning have typically operated with an initial gas pulse, and the density evolution was apparently determined by recycling from the flux conserver walls and the helicity injector. Since injecting helicity involves injecting magnetic flux and since plasma is frozen to magnetic flux, helicity injection tends to involve ingestion of cold particles. It is thus important to develop methods that inject helicity without excessive ingestion of cold particles. Coating the copper flux conserver and gun with a refractory metal (tungsten) layer, discharge cleaning, and titanium gettering have successfully controlled impurities.

FAST PARTICLES: UNDERSTAND EFFECTS OF FAST PARTICLES ON CURRENT DRIVE, STABILITY AND CONFINEMENT

Forming energetic particles in experiments and diagnosing their effects on stability and other physics have not been undertaken in spheromak experiments. There has been some computational modeling of orbits and preliminary scoping of the resulting current drive.

Research Requirements

Analysis of neutral beams to drive optimized current profiles is underway, but experiments will be needed to verify predictions before large-scale applications. As known from tokamak research, possible detrimental effects also need consideration. Energetic particles produced by neutral beams and fusion reactions can resonate with Alfvén eigenmodes and affect their stability. Relative to thermal particles, trajectories and confinement of energetic particles are much less sensitive to global magnetic fluctuations, but a quantitative assessment of the coupling is needed. Interaction of energetic particles with tearing instabilities is a new area of theoretical research and may have important, possibly beneficial, consequences.

RESISTIVE WALL MODES: DEMONSTRATE RESISTIVE WALL MODE CONTROL

The tilt and shift modes in existing spheromaks are controlled by flux-conserving walls, but for long-pulse or steady-state operation these instabilities will become resistive wall modes. No present facility is capable of addressing resistive wall physics in spheromaks.

Research Requirements

As in the RFP, macroscopic instabilities are sensitive to the location of a conducting wall and become violently unstable without a wall. The modes also are unstable when discharges exceed the time required for magnetic perturbations to diffuse through walls of finite conductivity. Effective feedback schemes for resistive walls and simultaneous, multiple MHD modes have been demonstrated experimentally on the Extrap-T2 and RFX-mod RFPs. This is encouraging for the spheromak, but an experimental confirmation of control is needed. The role of plasma rotation and rotational shear in stabilizing resistive wall modes in the spheromak also needs evaluation.

Incorporating thin-shell and external vacuum models is a tractable numerical approach for resistive wall studies, and simulation of feedback stabilization needs time-dependent sources. Damping for rotational stabilization may require sound wave damping or other kinetic effects.

TECHNOLOGY: DEVELOP THE TECHNOLOGY FOR LONG-PULSE OPERATION.

Technology development is presently done as needed for specific experiments, but will need to be addressed more systematically for future, fusion-level experiments.

Research Requirements

Helicity injection technology will need considerable development to handle steady-state conditions including heat loads if it continues to be used for spheromak sustainment. Current drive techniques are likely to have specific technology requirements that differ in part from those in tokamaks. Wall properties do not appear to limit performance at present, but are likely to need improvements to handle heat loads and other long-pulse issues. Options for divertors appear promising but need development. New facilities will be required. It is premature to define them at this time.

Diagnostics for CTs

As a result of low budgets for CT experiments, diagnostic capabilities have been limited and have constrained advances in understanding. It is important to improve the diagnostic capabilities of the larger FRC and spheromak experiments. The measurements should be nonperturbing. It is also important that research budgets recognize the need to fund the researchers who build, operate and interpret these diagnostics.

Theory and Modeling for CTs

Analytical theory provides an important basis for understanding CT physics. In cases where nonlinear effects and specific geometric considerations are important, numerical computation is required to solve theoretical models. Integrated computational simulations that model entire devices can be used in two ways. First, they provide time-dependent three-dimensional data that compensate practical limitations on laboratory diagnostics and help us understand the behavior of existing experiments. Second, they act as a predictive design tool where new ideas can be assessed prior to hardware construction or modification. Here, the objective is to lower the cost and shorten the time required to achieve the goals of the CT program. The appropriate validated simulation codes will need to integrate core plasma models with physical boundary effects, detailed geometry, and heating and current drive sources.

Reactor Considerations for CTs

Existing detailed reactor studies for CTs are more than two decades old and need updating using present methodology to include advances both in physics understanding and technology. In contrast to tokamak reactor studies (where the physics parameters are well known from decades of intensive research on many machines of widely varying sizes), CT reactor studies are needed to guide researchers regarding the physics parameters needed to make a viable reactor. It would be very useful to consider impacts on the possible reactor designs associated with the lack of toroidal magnets, high beta, and other features of the FRC and spheromak that are expected to improve reliability and ease of maintenance. This will also provide an opportunity to examine the possible use of advanced fuels in more detail than previous studies. This is especially important for the FRC as its high beta allows fusion power production at a minimum synchrotron radiation, making it a leading candidate for such fuels. Undertaking such studies with parameterized physics, e.g., of confinement, would fill an important need for CT research.

FRC ISSUES	**EXISTING CONCEPT-EXPLORATION FACILITIES**$^{(A)}$	**NEEDED FACILITIES**		**EXISTING THEORY, CODES**$^{(B)}$	**NEEDED THEORY, CODES**
		HIGH-*S* EXPERIMENT	**INTEGRATED PHYSICS EXP.**		
STABILITY AT LARGE *S*	Generate & sustain using RMF, merging-spheromaks, theta-pinches; divertors and field-shaping effects	20-50 mWb flux FRC: stability phys. (kinetic ions, shear flow, etc.)	Extend stability to higher-flux, larger size FRC	Nonlinear, resistive, & Hall MHD, hybrid; PIC, prolate & oblate geom.	Extend physics; verify & validate codes, explain & predict exp.
TRANSPORT	Transport in RMF-formed and sustained FRCs and in pulsed FRCs	Transp. in high-*s*, stable, quasi-stationary FRC; transp. at low v_d/v_i	Neutral-beam and/or rf heating, diagnostics for micro-instab.	PIC, Hamiltonian Symplectic	Model anomalous transport mechanisms in FRC
CURRENT DRIVE AND SUSTAINMENT	RMF driven; stabilization by RMF fields, pulsed FRCs	TBD: NBI, RMF or other means	Steady-state sustainment; diagn. for current driven micro-instab.	Nonlinear, resistive, & Hall MHD, hybrid; prolate & oblate geom., PIC, Hamiltonian Symplectic	Nonlinear, resistive, & Hall MHD, hybrid; prolate & oblate geom., PIC
FAST PARTICLES	Betatron orbit electrons	Neutral beam injection	Neutral beam injection	Hybrid codes	Hybrid or kinetic codes, PIC
HEATING	RMF, merging (reconnection), ohmic heating, theta pinch, compression	RMF, neutral beams, radiofrequency	RMF, neutral beams, radiofrequency, compression	Ohmic heating; test particle calculations for RMF stochastic heating	Nonlinear, resistive, & Hall MHD, hybrid codes with sources

(A) *Includes Colorado FRC, FRX-L, MRX, PFRC, PHD, PV Rotomak, SSX, TCS-U*

(B) *Includes HYM, NIMROD, Hi-Fi, RMF-Hamiltonian, Symplectic, PIC*

Table 3.

SPHEROMAK ISSUES	EXISTING CONCEPT-EXPLORATION FACILITIES(A)	NEEDED FACILITIES		EXISTING THEORY, CODES(B)	NEEDED THEORY, CODES
		SUST. & CONF. EXPERIMENT	INTEGRATED PHYSICS EXP.		
SUSTAINMENT AND CONFINEMENT	CW inductive helicity-injection, Hel. inj. data base, inductive drive, efficiency study	High-power & vac.; test concepts from CE exp., simulations	Extend best sustainment method to long-pulse, etc.	G-S fits to data, resistive MHD (rMHD) simulations	2-fluid simulation with kinetic effects; boundary cond. consistent with plasmas
FORMATION	CW inductive helicity injection, inductive, multi-pulsed startup, gun-plasma coupling efficiency	High-power & vac.; test concepts from CE exp., simulations	Extend best formation method to long-pulse, etc.	rMHD modeling, helicity balance	2-fluid simulation with kinetic effects; boundary cond. consistent with plasmas
TRANSPORT	Data base for electron confinement	Electron transport	Electron & ion transport; auxiliary heating; diagnostics	G-S fits to data, rMHD with collisional closure	Combined macro- & microscopic phys.; integrate with FSP
BETA LIMITS	Database on limiting beta, bow-tie geom. allows high Mercier β limit	Limits for various current drive methods	Find and study beta limits; heating, diagnostics	G-S fits to data, linear and nonlin. rMHD, Mercier limits	Nonlin. 2-fluid simulation with kinetic effects
PARTICLE BALANCE AND DENSITY CONTROL	Wall conditioning & gas injection, flows in helicity injection	N/A	Wall cond., particle injection, dedicated exp.	N/A	Integrate atomic physics and coupling to wall and other sources into codes
FAST PARTICLES	N/A	N/A	Neutral beam injection	delta-f treatment	Adapt energetic particle modeling
RESISTIVE WALL MODES	N/A	N/A	Based on RFP & tokamak experience	N/A	Extend RFP and tokamak analyses to spheromak

(A) *Includes CalTech, DRX, HIT-SI MRX, PBX, SSPX (database only), SSX*

(B) *Includes Corsica (Grad.-Shaf. based), NIMROD, Hi-Fi*

Table 4.

Connections Between Research Requirements and ReNeW Research Thrusts

Three primary research thrusts (16-18) emerge from the Optimizing the Magnetic Configuration Theme. These thrusts describe configuration optimization associated with major variables of aspect ratio, 3-D shaping, and degree of external magnetization. They have been designed to expose the opportunities and research actions for the development of the integrated approach to fusion provided by each configuration, plus the scientific value of linked multi-configuration research.

All of the research thrusts in ReNeW concern toroidal confinement and are therefore relevant to some degree to tokamak and alternates alike. Table 5 provides a sense of the connections between the research needs for toroidal alternates and the science appearing in all 18 thrusts.

3 = thrust addresses primary research need(s); configuration is a primary contributor.

2 = thrust addresses significant research need; configuration is a major contributor.

1 = configuration is a potential contributor and/or benefactor of thrust.

RESEARCH THRUST	Stell.	ST	RFP	CT
1. Develop measurement techniques to understand and control burning plasmas.	2	2	2	1
2. Control transient events in burning plasmas.	3	2	1	1
3. Understand the role of alpha particles in burning plasmas.	1	2	1	1
4. Qualify operational scenarios and the supporting physics basis for ITER.	1	2	1	1
5. Expand the limits for controlling and sustaining fusion plasmas.	1	2	2	2
6. Develop predictive models for fusion plasmas, supported by theory and challenged with experimental measurement.	3	3	3	3
7. Exploit high-temperature superconductors and other magnet innovations to advance fusion research.	2	2	1	1
8. Understand the highly integrated dynamics of dominantly self-heated and self-sustained burning plasmas.	2	2	1	1
9. Unfold the physics of boundary layer plasmas.	1	2	1	1
10. Decode and advance the science and technology of plasma-surface interactions.	1	2	1	1
11. Improve power handling through engineering innovation.	2	3	2	2
12. Demonstrate an integrated solution for plasma-material interfaces compatible with an optimized core plasma.	1	2	1	1
13. Establish the science and technology for fusion power extraction and tritium sustainability.	1	1	1	1
14. Develop the material science and technology needed to harness fusion power.	1	1	1	1
15. Create integrated designs and models for attractive fusion power systems.	1	2	1	1
16. Develop the spherical torus to advance fusion nuclear science.	2	3	2	2
17. Optimize steady-state disruption-free toroidal confinement using 3-D magnetic shaping, and emphasizing quasi-symmetry principles.	3	2	2	2
18. Achieve high-performance toroidal confinement using minimal externally applied magnetic field.	1	2	3	3

Table 5

OPTIMIZING THE MAGNETIC CONFIGURATION THEME MEMBERS

STELLARATOR PANEL

DAVID ANDERSON, University of Wisconsin-Madison (Panel Leader)
JEFF FREIDBERG, Massachusetts Institute of Technology
JEFF HARRIS, Oak Ridge National Laboratory
CHRIS HEGNA, University of Wisconsin-Madison
STEPHEN KNOWLTON, Auburn University
MICHAEL MAUEL, Columbia University
ALLAN REIMAN, Princeton Plasma Physics Laboratory
ANDREW WARE, University of Montana
HAROLD WEITZNER, New York University

SPHERICAL TORUS PANEL

STEVEN SABBAGH, Columbia University (Panel Leader)
NIKOLAI GORELENKOV, Princeton Plasma Physics Laboratory
CHRIS HEGNA, University of Wisconsin-Madison
MICHAEL KOTSCHENREUTHER, The University of Texas at Austin
DICK MAJESKI, Princeton Plasma Physics Laboratory
JON MENARD, Princeton Plasma Physics Laboratory
MARTIN PENG, Oak Ridge National Laboratory
AARON SONTAG, Oak Ridge National Laboratory
VLAD SOUKHANOVSKII, Lawrence Livermore National Laboratory
DANIEL STUTMAN, Johns Hopkins University

REVERSED FIELD PINCH PANEL

HANTAO JI, Princeton Plasma Physics Laboratory (Panel Leader)
BRETT CHAPMAN, University of Wisconsin-Madison
DANIEL DEN HARTOG, University of Wisconsin-Madison
GENNADY FIKSEL, University of Wisconsin-Madison
PIERO MARTIN, Conzorzio RFX
FARROKH NAJMABADI, University of California-San Diego
STEWART PRAGER, Princeton Plasma Physics Laboratory
CARL SOVINEC, University of Wisconsin-Madison

COMPACT TORUS PANEL

BICK HOOPER, Lawrence Livermore National Laboratory (Panel Leader)
ELENA BELOVA, Princeton Plasma Physics Laboratory
MICHAEL BROWN, Swarthmore College
SAM COHEN, Princeton Plasma Physics Laboratory
ALAN HOFFMAN, University of Washington-Seattle
SCOTT HSU, Los Alamos National Laboratory
THOMAS INTRATOR, Los Alamos National Laboratory
THOMAS JARBOE, University of Washington-Seattle
RICHARD MILROY, University of Washington-Seattle
MICHAEL SCHAFFER, General Atomics
CARL SOVINEC, University of Wisconsin-Madison
XIANZHU TANG, Los Alamos National Laboratory
SIMON WOODRUFF, Woodruff Scientific, Inc.
MASAAKI YAMADA, Princeton Plasma Physics Laboratory

RESEARCH THRUST COORDINATORS

Thrust 16 – Develop the spherical torus to advance fusion nuclear science.
DICK MAJESKI, Princeton Plasma Physics Laboratory
Thrust 17 – Optimize steady-state disruption-free toroidal confinement using 3-D magnetic shaping, and emphasizing quasi-symmetry principles.
STEPHEN KNOWLTON, Auburn University
Thrust 18 – Achieve high-performance toroidal confinement using minimal externally applied magnetic field.
CARL SOVINEC, University of Wisconsin-Madison

THEME LEADERS

JOHN SARFF, University of Wisconsin-Madison, Chair
MICHAEL ZARNSTORFF, Princeton Plasma Physics Laboratory, Vice Chair
SAM BARISH, Office of Fusion Energy Sciences, US Department of Energy

PART II:
RESEARCH THRUSTS

PART II: RESEARCH THRUSTS

Introduction

The thrusts described in the following chapters are the main results of the Workshop. They constitute eighteen concerted research actions to address the key challenges of the fusion quest. Each thrust attacks a related set of fusion science issues, using a combination of new and existing tools, in an integrated manner.

The thrusts range in content over all the issues delineated in the five themes. They also include a wide range of sizes, from relatively small, focused activities to much larger, broadly encompassing efforts. This spectrum should enhance the flexibility of OFES planning.

Three elements characterize, in varying degrees, all the ReNeW thrusts:

- Advance in fundamental science and technology, such as the development of broadly applicable theoretical and simulation tools, or frontier studies in materials physics.
- Confrontation with critical fusion challenges, such as plasma-wall interactions, or the control of transient plasma events.
- The potential for major transformation of the program, such as altering the vision of a future fusion reactor, or shortening the time scale for fusion's realization.

Thrust organization: The thrusts are organized into theme chapters; each thrust is identified with the theme that contains its most central issues. This organization mirrors that of Part I, and is convenient for similar reasons. However, we emphasize again that the content of a typical thrust transcends that of any single theme. The scientists contributing to most thrusts are from a variety of research areas, and key elements of a given thrust may stem from ideas developed in several themes.

Thrust summaries: The presentation of each thrust begins with a one-page summary. These summaries are intended to provide a quick sense of what the thrust is about: the science and technology questions it addresses, with context, and the major actions proposed. We emphasize that the one-page summaries do not include all features and activities of the thrust — information that will be found in the more detailed thrust description.

Linkages among thrusts: While each thrust attempts a certain stand-alone integrity, all are linked in various ways. In particular, the activities of one thrust might depend upon information to be discovered by another thrust. Such linkages are discussed explicitly in the following chapters.

Thrust 1: Develop measurement techniques to understand and control burning plasmas

The impressive progress in fusion science understanding has been enabled by the development of an extensive suite of advanced measurement techniques. Such measurements have also been instrumental in achieving active control of high-performance plasmas. As we move toward the "burning plasma era" of ITER and beyond to DEMO, the instruments making these measurements will be exposed to an increasingly hostile environment, and many existing techniques will be severely challenged and risk potential failure. The fusion science community must immediately begin the development of new and improved diagnostic systems to guarantee availability of the measurements essential for advanced plasma control and improved understanding of steady-state burning plasmas.

Key Issues:

- Developing diagnostics critical to the burning plasma research goals:
    - That provide the essential tools for scientific discovery in burning plasmas and reliably support real-time plasma control and machine protection.
    - That survive the hostile, long-pulse environment of ITER, and the even harsher, steady-state environment of a DEMO device.
- Developing techniques to measure uniquely important properties of burning plasmas, such as the behavior of confined fusion alpha-particles.

Key Questions:

- *What creative, new measurement techniques can fill acknowledged "measurement gaps" in present ITER plans?*
- *What opportunities for new measurement techniques arise through operation at high-performance, burning plasma parameters?*
- *What technological advances would mitigate the known risks and optimize measurement reliability in the harsh ITER environment?*
- *What are the minimum measurement requirements for DEMO, and what techniques can be developed to satisfy them in a DEMO environment?*

The US ITER Project is currently focused on delivering "in-kind" contributions of a few diagnostic systems based on existing measurement techniques. However, the US diagnostic community could also contribute to these and to a much broader range of essential measurement needs through a new, targeted burning plasma diagnostic development program. Targeting would be coordinated with efforts elsewhere, with preference for developments supporting US burning plasma research interests. This is a high-leverage way to maximize the US investment in ITER research and to contribute substantially to preparations for DEMO.

Proposed Actions:

- Prioritize burning plasma measurement needs, including those for DEMO.
- Perform phased developments targeted to high-priority needs, including prototyping on operating devices.
- Evaluate the success of these developments, and where appropriate, work to transfer techniques to ITER or other burning plasma devices.

Scientific and Technical Research

Measurements are key to scientific understanding. There are many examples of important strides in fusion science that were triggered by new measurements. New phenomena appear whenever measurement resolution is enhanced or an additional parameter is measured. The richness of plasma phenomena means that optimization of fusion performance is a complex process, which often draws on unexpected parts of the measurement tool kit. While the tool kit planned for ITER is similar, in many ways, to that for existing tokamaks, the applicability of many of these tools to a long-pulse, nuclear environment has yet to be established. Designs are not mature, and tradeoffs between capability and reliability have not been made. Finding robust alternate techniques for at-risk ITER measurements may be pivotal in achieving optimal ITER performance. For example, moving away from reliance on optical techniques, vulnerable due to degradation of plasma facing mirrors, toward more compatible microwave methods or X-ray techniques would be appropriate strategies. As currently envisioned, maintenance for many ITER diagnostic signal-gathering components involves robotic replacement of port plugs weighing up to 50 tons. These plugs (Figure 1) house the diagnostic "front-end" components (such as optical elements like mirrors) and shield the magnets and other subsystems from the fusion core. To deploy these components, innovative engineering strategies are required that more easily accommodate maintenance and thus improve overall reliability. For burning plasmas beyond ITER, the significantly harsher environment, the decreased diagnostic access, and the requirements of increased reliability make measurements on DEMO-like plasmas even more difficult and uncertain.

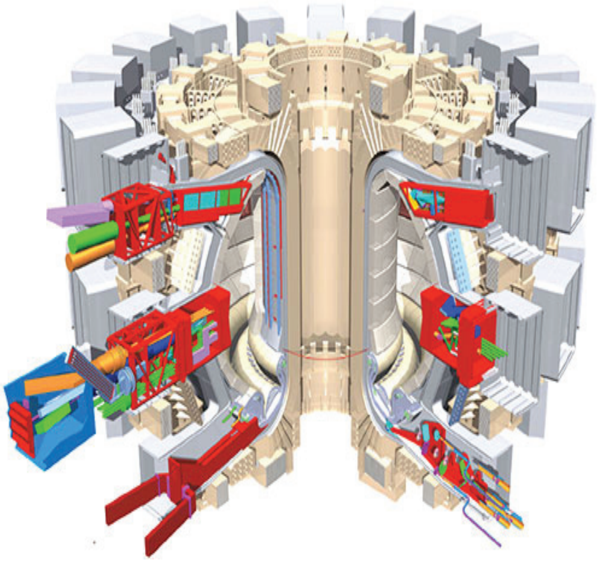

Figure 1. ITER port plugs (red) showing "front-end" diagnostic components.

GROUP 1A MEASUREMENTS FOR MACHINE PROTECTION AND BASIC CONTROL	GROUP 1B MEASUREMENTS FOR ADVANCED CONTROL	GROUP 2 PERFORMANCE EVALUATION AND PHYSICS
Plasma shape and position, separatrix-wall gaps, gap between separatrices Plasma current, $q(a)$, $q(95\%)$ Loop voltage Fusion power $\beta_N = \beta_{tor}(aB/I)$ Line-averaged electron density* Impurity and D, T influx (divertor*, & main plasma) Surface temperature (divertor & upper plates)* Surface temperature (first wall) Runaway electrons Halo currents Radiated power (main plasma, X-pt & divertor) Divertor detachment indicator (J_{sat}, n_e, T_e at divertor plate) Disruption precursors (locked modes, m=2) H/L mode indicator* Z_{eff} (line-averaged) n_T/n_D in plasma core ELMs Gas pressure (divertor & duct) Gas composition (divertor & duct)* Dust	Neutron and a-source profile Helium density profile (core) Plasma rotation (toroidal and poloidal)* Current density profile (q-profile)* Electron temperature profile (core)* Electron density profile (core and edge)* Ion temperature profile (core)* Radiation power profile (core, X-point & divertor) Z_{eff} profile Helium density (divertor)* Heat deposition profile (divertor)* Ionization front position in divertor Impurity density profiles* Neutral density between plasma and first wall n_e of divertor plasma* T_e of divertor plasma a-particle loss Low m/n MHD activity Sawteeth Net erosion (divertor plate) Neutron fluence	Confined a-particles TAE modes, fishbones* T_e profile (edge) n_e, T_e profiles (X-point) T_i in divertor Plasma flow (divertor) $n_T/n_D/n_H$ (edge) $n_T/n_D/n_H$ (divertor) T_e fluctuations n_e fluctuations* Radial electric field and field fluctuations Edge turbulence MHD activity in plasma core

Color coding: Expect performance to meet measurement requirements; maybe/maybe not; expect not to meet requirements.

Indicates at least one primary technique at risk due to mirror degradation.

** Indicates measurement for which US ITER Project has responsibility to provide a primary diagnostic.*

Table 1. ITER measurements categorized by role, performance expectation, mirror risk, and US ITER Project involvement.

Why a New Burning Plasma Measurement Program?

The understanding and control of fusion plasma behavior relies on a comprehensive set of measurements. Table 1 shows the diagnostic set currently planned for ITER. A broad variety of issues challenge designers in providing these needed measurements:

1) There are "acknowledged gaps," i.e., cases where it is recognized that no compatible techniques exist to measure a needed plasma parameter. An example is the lack of a qualified technique for the measurement of lost alpha particles.

2) Some presently planned ITER diagnostics are vulnerable due to potential failure of near-plasma components. An example is degradation of plasma facing diagnostic mirrors due to plasma-induced deposition or erosion. Failure of such "at-risk" systems could effectively create more "gaps" in the measurement capability, leading ultimately to reduced effectiveness of the research program. ITER measurements, indicated by brown shading in Table 1, carry this risk. It would be prudent to develop robust methods to supplement or replace "at-risk" diagnostics.

3) As planning matures for the research program, new measurement needs are sure to arise, and "emerging gaps" will be identified. An example might be an added measurement needed for a promising control scenario successfully developed on an operating device.

4) To avoid difficult maintenance and repair procedures involving remote robotic handling, failure rates much lower than on present-day devices must be achieved. This represents a significant engineering and technological challenge affecting a variety of diagnostic components, including maintenance, alignment, and calibration subsystems.

5) It is highly likely that some techniques yielding crucial measurements will not work in a DEMO environment, e.g., proximity-coil-based magnetic measurements and optical-based measurements. Thus, new techniques are needed for these measurements at risk in DEMO. In particular, novel and robust plasma-diagnostic interfaces should be developed.

As a high-priority US diagnostic activity, the US ITER Project Office is delivering a specific subset of diagnostics to the ITER Organization. US developers will have to demonstrate, through qualification testing, that this hardware will operate reliably in an ITER environment. Thus, some of the issues mentioned earlier will be addressed for these specific diagnostics. However, the USIPO responsibility does not extend to issues 1), 3), and 5). Nor is it likely to be able to fully address generic aspects of 2) or 4), such as first-mirror lifetime prediction and validation. US expertise could help address at least some of these areas not included in the USIPO scope. Unfortunately, as presently structured, no US program exists to fund such efforts.

Major US experimental fusion programs have benefited greatly from Office of Fusion Energy Sciences (OFES) awards for "Development of Diagnostic Systems for Magnetic Fusion Energy Sciences Experiments." However, this program supports developments benefiting research on *existing facilities* only. Major US fusion programs also support valuable diagnostic developments in support of associated research. In both cases, developments may be applicable to burning plasmas. However, the considerable effort required to translate these developments into conceptual ITER-relevant or DEMO-relevant designs and to evaluate expected performance at burning plasma parameters presently has no funding path available.

A broader mandate is needed for US experts to consider a full range of measurement issues. For example, robust, burning-plasma-relevant alternatives for techniques that work well on existing devices must be developed. Generic efforts to reduce risk and enhance reliability in a burning plasma environment are needed. A targeted solicitation based on prioritized issues could be used to launch a program to develop burning plasma diagnostics outside of the scope of the USIPO, in coordination with efforts worldwide. Dividends from this new investment would be a more vigorous research program on ITER and a firmer basis for the design of measurements for next-step steady-state devices.

ITER provides a near-term focus for this Thrust. Implementation of diagnostics on DEMO is longer term, but also a very critical issue. In most cases for DEMO, measurement concepts do not exist. Fundamental materials issues will confront designers of sensors in proximity to the plasma. All techniques will need to adapt to greatly reduced access through which to collect signals. The envisioned effort, to develop reliable techniques compatible with the more hostile steady-state DEMO environment, demands that this development should also begin immediately.

Summary of Actions Proposed for this Thrust

Actions in this Thrust go significantly beyond US activities in this area in recent years. The first could be requested from a US panel of experts and could begin immediately:

> Prioritize burning plasma measurement needs, including those for DEMO.

This prioritization could be repeated periodically as new information on plasma behavior is gathered from operating devices, as the US role in burning plasma research becomes better defined, and as diagnostic design activities bring measurement issues into sharper focus. Following a periodic OFES solicitation process, the diagnostic development community will be involved in two critical activities:

> Perform phased developments targeted to high-priority needs, including prototyping on operating devices.
>
> Evaluate the success of the developments, and where appropriate, support the transfer of techniques to ITER or other burning plasma devices.

Thrust Elements

Prioritize burning plasma measurement needs, including those for DEMO.

A first step in this Thrust is to establish development targets. This is a new regime of diagnostic development. The requirements and risks are well described in Chapter 12 of the "Special Issue on Plasma Diagnostics for Magnetic Fusion Research," *Fusion Science and Technology*, **55**, Feb. 2008. As a first step toward identifying high-priority burning plasma measurement needs, a US panel of experts could undertake a review of the ITER measurement requirements and the corresponding expected capabilities. The panel could evaluate the acknowledged and likely emerging gaps, the "at-risk" systems to target for the supplemental system development, and how these issues relate to the high-priority US research interests.

The ITER Organization is presently working with the ITER parties to establish the "functional specifications" for the planned diagnostics, which will provide the goals for the detailed design of these systems. These specifications, along with basic layout designs, expected system performance, and environmental risks, will be presented for each of the planned systems. A significant body of information will thus be available in the near future, which, upon thoughtful review, could inform the US targeting effort. It will be beneficial to work with the ITER Organization and other stakeholders in this review process.

An important deliverable in the review proposed above could be a prioritized set of burning plasma measurement issues to form the basis of a solicitation for a US development effort. Clearly addressing acknowledged or newly identified measurement gaps could be one class of such issues; improving the capability or reliability of presently planned ITER measurements could be another. Improvement of systems not presently allocated to the US could also be considered. Development of a new class of more robust diagnostics to supplement high-risk systems would also be important and would address concerns related to DEMO. A related set of topics includes better definition of the measurement needs for DEMO, and an exploration of the feasibility of measurement concepts for such devices. Priority could be given to topics that US experts can plausibly address, that have high impact, that benefit US research interests, and that are not duplicated elsewhere. As planning for research on ITER evolves and the US role becomes better defined, it is likely that measurement priorities will be refined. In addition, as designs for the presently planned and credited ITER diagnostics become more mature, measurement issues will come into sharper focus. Thus it would be advantageous to repeat this prioritization periodically to adapt to changing needs.

> ***Perform phased developments targeted to high-priority needs, including prototyping on operating devices.***

A new burning plasma diagnostic development program could be launched, dedicated to US developments addressing high-priority burning plasma measurement issues. The new program could be phased, with early success leading to funding renewal. The initial awards could be for feasibility studies, with success leading to awards for R&D and design of prototype diagnostics on suitable existing facilities and, finally, for the actual implementation and demonstration of the prototype. Stimulating the creative talent in the US to address these measurement needs should be a major priority for the US Fusion Energy Sciences program. Similar burning plasma diagnostic development programs now exist in other countries, for example, Europe (European Fusion Development Agreement Diagnostics Work Programme) and Japan (Advanced Diagnostics for Burning Plasma Experiment). A similar program has also been proposed in Australia. A coordinated approach with these programs would be essential to maximize the benefit to ITER and future burning plasma devices.

> ***Evaluate the success of the developments and, where appropriate, work with the ITER Project to implement qualified techniques.***

After a new technique is successfully qualified, a handoff to ITER for full implementation is a critical step. Over the last several years, the ITER Project has been receptive to incorporation of properly qualified new ideas. It is likely that diagnostics will be installed in several phases, between operational cycles, permitting flexibility to introduce new systems. Most diagnostics are housed in "port plugs," which are replaceable periodically throughout the lifetime of ITER. Communication with the ITER Project is key to successful implementation, and promising new developments will need to be promoted as early as possible. Communication with other Domestic Agencies would be needed for proposed modifications to credited systems not supplied by the US. It should be noted that ITER will serve as an excellent test bed for robust DEMO-relevant diagnostics.

Scale of Effort and Readiness for Thrust

To fully exploit US diagnostic expertise to address burning plasma measurement issues, significant resources are needed. After issues are identified and prioritized, a reasonable number of feasibility studies, in the range of ten, could be pursued. A level of support comparable to the present OFES development program may be adequate for this early phase. Later phases of the awards would require more resources per award and fewer awards to support R&D, design activities, and, finally, prototyping on operating devices. Depending on community response, a reasonable level for later phases could be several times that of the present program. Once fully implemented, this level of funding would provide for a mix of new feasibility studies along with more mature hardware design and prototyping awards. It should also be noted that access to operating US facilities and run time would be required during the prototyping phase.

In recent years, as the US community began to focus on ITER diagnostics, ideas for new measurements arose. With a clear funding path, some of these concepts could be the kernels of proposals for new burning plasma measurements. The time scale for development of new techniques, from inception to routine operation, is typically five to ten years. On the other hand, implementing a new idea for maintenance, robustness, or calibration could occur on a substantially shorter time scale and potentially be incorporated into ongoing ITER designs. This requires that the development activity start in the near term (< 2 years). By incorporating minor modifications to "front-end" components, some systems could be used for multiple measurements, and there are good ideas to more fully exploit presently credited systems. Such expansion of capability could be easily implemented if modifications are defined early in the development process. As described, to most positively have an impact on ITER and get the best return on investment, developments should start as soon as possible.

Examples of candidate developments include advanced polarimetry and reflectometry techniques for local B-field measurements, microwave scattering for density fluctuation measurements, microwave techniques for fuel-mix measurements, optical Penning gauge measurements of neutral species in the divertor, methods to monitor and clean optical mirrors, erosion and retention measurements, remote X-ray sensing for plasma control on DEMO, and robust in-situ, real-time calibration techniques.

Although developments for ITER require the highest priority, support of measurements for DEMO should not be ignored. The environment near the plasma will be much more hostile, and even higher reliability will be needed. Initially, engineering will be needed in the areas of diagnostic integration with the device and studies of environmental impacts. Later, however, work in preparing prototype instruments for test on ITER will be required.

Integration of Thrust Elements

The elements of this Thrust form a coherent program aimed at leading measurement science and engineering into the burning plasma era. After reviewing ITER measurement capabilities and risks, an expert panel will define a set of actionable issues. To address these issues, measurement experts develop and qualify new techniques, which, if successful and applicable, are presented for implementation. Coordination will be beneficial at all stages between those involved in this Thrust and other diagnostic groups concerned with burning plasmas — most importantly with the ITER Diagnostic Division and the diagnostic management groups representing the other domestic agencies. This coordination could be fostered by including experts from these groups in the US expert panels involved in choosing high-priority issues and evaluating the potential of new techniques. Such groups include the Diagnostics Topical Groups of the US Burning Plasma Organization (and corresponding groups in other nations) and the International Tokamak Physics Activity. Similarly, there could be coordination with the independent burning plasma diagnostic development organizations in Europe, Japan, and elsewhere.

Relation to Other Thrusts and Other Scientific Benefits

This Thrust is focused on the development of diagnostics essential to the control and scientific mission of burning plasmas. There are important synergies between this Thrust and several other burning plasma thrusts in this document. Measurement needs in other burning plasma thrusts could have an impact on the identification of high-priority issues for this Thrust's development phase and will likely benefit from successful implementation of new techniques. Examples are not hard to imagine. The adequacy of measurements needed to assess core and edge stability could be important in Thrust 2 for the control of transient events. The capability of planned measurements of alpha-particle distributions, alpha-particle instabilities, and alpha-particle losses could be carefully considered relative to the needs of Thrust 3 for the understanding of alpha physics. As burning plasma operational scenarios are qualified in Thrust 4, critical measurement needs could be identified and developed in existing devices, which are outside of those presently planned for ITER. Critical control solutions explored in Thrust 5 could depend on development of sensor capabilities beyond those presently planned. As models to predict plasma behavior in Theme 6 are tested on existing devices, validating measurements could be used that exceed the capabilities of planned ITER diagnostics and that could be developed for burning-plasma compatibility in this Thrust. New measurements could be needed for control of noninductive, self-sustaining plasmas explored in Theme 8. Facilities developed in Thrust 8 could also be used as test beds for DEMO diagnostics.

In summary, with US research interests in mind, the proposed measurement Thrust will engage US experts to prioritize measurement issues for ITER and DEMO and then develop, qualify, and implement new techniques to address these issues. Diagnostic development is an area that traditionally nurtures young talent. The Thrust will provide a valuable opportunity for young scientists and engineers to engage in high-impact burning plasma issues now. Such development is excellent insurance for a vigorous US ITER research participation, and an important prerequisite for meaningful planning for DEMO.

Thrust 2: Control transient events in burning plasmas

Transient events such as disruptions and edge localized modes (ELMs) cause high peak heat loads on plasma facing surfaces, potentially leading to rapid erosion or melting. Disruptions also have the potential to cause damage through large electromagnetic forces and intense beams of high-energy electrons. Although ELMs and disruptions are generally tolerated in present tokamaks, the consequences will be much more severe in future burning plasmas because of the larger thermal and magnetic energies and longer pulse lengths; such events will reduce operational availability and shorten the lifetime of plasma facing components. It is vital to minimize such events in ITER and to mitigate their consequences when they occur.

Key Issues:

- Characterization of disruptions. *What are the causes of disruptions in present facilities? Do they always have detectable precursors? How do the consequences of disruptions extrapolate to ITER and other burning plasmas?*
- Capability to predict disruptions. *Can plasma stability be assessed accurately enough in real time to predict stability limits? Can disruption precursors be reliably detected?*
- Capability to avoid disruptions. *If disruptions can be predicted, what remedial actions can be taken to suppress the instability or move the discharge to a more stable operating point?*
- Means to minimize the impact of disruptions. *What is the best means to mitigate the effects of disruptions in ITER? What are the consequences of a mitigated disruption?*
- Means of robustly avoiding or suppressing ELMs. *Can ELMs be reliably eliminated or their impulsive power loading sufficiently reduced through 3-D magnetic fields and other means of modifying the plasma edge? How do ELM avoidance techniques extrapolate to ITER?*

Proposed Actions:

- Characterize the causes and consequences of disruptions in existing facilities. Benchmark predictive models of disruptions and disruption mitigation against existing data.
- Use existing tokamaks to develop and test tools for real-time prediction and measurement of plasma stability, and detection of disruption precursors.
- Use existing tokamaks to develop techniques for suppressing instabilities or steering the operating point away from stability limits. Demonstrate these techniques in longer pulses on the emerging generation of superconducting devices.
- Assess strategies for mitigating disruptions through a rapid but benign shutdown of the discharge. Demonstrate solutions in medium and large tokamaks for extrapolation to ITER.
- Develop techniques to mitigate ELMs through modification of edge plasma transport and stability. Demonstrate the solutions in medium and large tokamaks for extrapolation to ITER.

The US fusion program is well positioned to carry out much of the required work in existing tokamaks, with modest upgrades to diagnostics and auxiliary systems, but substantial increases in experimental time and human resources. Further technology development will be required to extend these techniques to the burning plasma regime in ITER.

Research Elements

Transient events such as disruptions and ELMs must be minimized or eliminated in a burning plasma. Since a disruption is a singular event that terminates the plasma pulse, the conditions that generate a disruption must be predicted and avoided. Techniques to mitigate the effects of a disruption must also be available, but should only be required in rare instances if avoidance fails. The transient power loading due to ELMs must also be avoided in burning plasmas. This impulsive ELM-induced transport is the norm for steady high-confinement mode (H-mode) operation in present devices, but a burning plasma requires a more continuous process that still allows the optimum level of heat and particle transport in the plasma edge.

Disruptions and ELMs comprise distinct categories of off-normal transient events and have distinct research requirements. Research in this area can be organized into four broad elements:

- Prediction of disruptions.
- Avoidance of disruptions.
- Mitigation of disruptions.
- Avoidance of ELM-induced impulsive power loads.

Many of the tools to achieve these goals are now being developed, and concepts exist for others. In the next 10 to 20 years, these building blocks must be developed, and integrated into systems that will ensure reliable, near-steady-state operation of magnetically confined fusion plasmas.

Prediction of disruptions

Sustained, full-performance operation of a burning plasma requires that the conditions leading to a disruption be identified in time to take action to avoid or mitigate the disruption, a challenging problem even in existing tokamaks. Accurate and reliable prediction is necessary to maximize fusion performance without exceeding stability limits, while minimizing "false positive" results that may lead to unnecessary retreat from high-performance conditions or shutdown of the discharge. Several levels of prediction must be developed, including empirical characterization of operating limits, real-time assessment of the plasma operating state including calculation of magnetohydrodynamic (MHD) stability limits, and identification of plasma "symptoms" indicating that a disruption could ensue. Reliability requirements motivate a broad portfolio of prediction methods, and redundancy of detection systems.

Key research steps include:

- Characterization of disruptions in existing data: cause of disruptions and their relative frequency, identifiable precursors, electromagnetic and thermal loads.

- Development and benchmarking of 2-D and 3-D models for disruption dynamics, including electromagnetic and thermal loads, runaway electrons, wall interaction, etc.
- Theoretical and numerical stability modeling, including time-dependent scenario modeling, to improve capabilities for disruption prediction.
- Development and benchmarking of real-time energy balance and transport analysis, for early warning of impurity accumulation and other possible disruption precursors.
- Development of real-time stability calculations, to warn of proximity to stability limits.
- Development of direct, real-time determination of plasma stability through "active MHD spectroscopy" (MHD damping rate measurement by exciting the mode at low amplitude).
- Development of diagnostics and real-time analysis for identification of a growing instability at amplitude well below the threshold for disruption.
- Development and testing in present devices of sensors that can provide the required measurements for disruption prediction in a long-pulse, nuclear environment.

Avoidance of disruptions

Disruption avoidance includes both passive and active techniques to avoid instabilities. The rapid growth rate of many ideal MHD instabilities means that their stability limits must be detected and avoided well ahead of the onset. On the other hand, normal operation may lie beyond passive stability limits such as those for the neoclassical tearing mode or the resistive wall kink mode; these cases require active stabilization of the plasma. The disruption predictors described in the preceding list form the input to these passive and active controls.

Key research steps include:

- Modeling and experimental benchmarking of control strategies to steer the operating point away from an impending instability, without approaching other operating limits.
- Modeling and experimental benchmarking of control strategies to recover normal operation in the event of an instability or another potential disruption-inducing event.
- Modeling and experimental benchmarking of strategies for recovery of normal operation in the event of an off-normal condition caused by component failure or human error, and specification of operating practices that minimize such events.
- Development of actuators capable of modifying the pressure, current density, and rotation profile with minimal auxiliary power, and suitable for use in a nuclear environment.
- Modeling and experimental benchmarking of active stability control using actuators such as localized current drive and non-axisymmetric coils.
- Development of high-bandwidth coils for MHD spectroscopy and active feedback control, suitable for use in a nuclear environment.
- Assessment of the impact of implementing disruption prediction and avoidance techniques, consistent with ITER and DEMO requirements on fusion power production.

Mitigation of disruptions

In the event that normal operation cannot be maintained or recovered by these techniques , suitable actions must be available to avoid the worst effects of a disruption. The most desirable solution is a controlled shutdown of the plasma, in which the thermal energy and plasma current are brought smoothly to zero. As a last resort, a rapid shutdown must be available, for example, by gas or pellet injection. Massive gas injection has successfully mitigated the electromagnetic and thermal loads on internal components in current experiments. The primary unsolved issue is the possible generation of a runaway electron avalanche during a disruption or rapid shutdown, although issues also remain for the extrapolated electromagnetic and thermal loads on ITER and DEMO. Reliability requirements motivate the development of a broad portfolio of mitigation strategies, and redundancy of hardware systems. Detailed, validated models are also crucial for extrapolation of mitigation techniques to ITER, where timely progress in the research program depends on having only a small number of unmitigated disruptions.

Key research steps include:

- Rapid and reliable disruption prediction, enabling a control decision to abandon disruption avoidance and initiate shutdown.
- Development of gas, solid, or liquid injection systems that deliver a sufficient quantity of electrons to the plasma core rapidly enough for collisional suppression of runaways.
- Development of alternate solutions for runaway electron suppression (e.g., stochastic magnetic fields, or control of the runaway beam long enough to decelerate it).
- Development and benchmarking of 2-D and 3-D models for the entire shutdown process: mass delivery and transport to the plasma core, the resulting shutdown of the discharge, and accompanying generation, confinement, and loss of runaway electrons.

Avoidance of ELM-induced impulsive heat loads

The most desirable solution for avoidance of ELMs is one that maintains a high H-mode pedestal pressure at the edge of the plasma, without the edge localized modes that are often driven by the strong edge gradients, but with sufficient transport to avoid buildup of energy and particles. Several operating regimes with no ELMs or very small ELMs have been identified, and non-axisymmetric (3-D) magnetic fields have been demonstrated to suppress ELMs. However, the physics of these processes is not yet well understood. Uncertainties in extrapolation to burning plasma regimes and the possibility that close-fitting 3-D magnet coils may not be credible in a DEMO device motivate the development of multiple methods of ELM control.

Key research steps include:

- Develop predictive capability for the effects of 3-D stochastic magnetic fields on particle transport and thermal transport in the plasma edge.
- Determine requirements for the 3-D magnetic spectrum, including radial localization, ELM suppression over a range of edge q values, and minimization of deleterious nonresonant fields.

- Evaluate the effectiveness of fueling and pumping — including core pellet fueling — in the presence of 3-D fields used for pedestal density reduction and ELM control.

- Identify the underlying mechanisms responsible for the modification of edge transport and stability in ELM-free regimes such as the quiescent H-mode (QH mode) and the enhanced D-alpha (EDA) H-mode.

- Identify and test other means of edge profile control — e.g., shallow pellet injection, radiofrequency-based methods, recycling control (lithium wall), or external rotation shear modification.

- In improved confinement regimes with a low-confinement mode (L-mode) edge, assess whether the reduced edge pressure gradient is compatible with the required high confinement, and assess any possible reductions in the global stability limits and achievable bootstrap fraction.

- Identify and test ELM-tolerant wall concepts — e.g., liquid walls.

Scale of effort

A US research thrust to establish the basis for reliable operation, free of ELMs and disruptions, could make a major contribution to the world fusion program. The output from this research thrust would include:

- Development of MHD stability measurements and calculations for real-time execution, as well as detailed, physics-based models of disruptions, ELMs, and their control.

- Incorporation of validated models into simulations suitable for prediction, avoidance, and mitigation of transient events in ITER and future burning plasmas.

- Development and assessment of solutions for detection, avoidance, and mitigation of transient events, including diagnostics and actuators suitable for burning plasmas.

- Implementation of the techniques of disruption prediction, avoidance, and mitigation in ITER and assessment of their effectiveness.

- Confidence that a sufficiently low rate of transient events can be routinely achieved in future devices.

Virtually all of the research needed for disruption prediction and ELM control, and a large proportion of the research for disruption avoidance and disruption mitigation, can and should be done in **existing short-pulse facilities** (C-Mod, DIII-D, and NSTX in the US, plus international facilities such as JET and ASDEX-Upgrade). At most, modest upgrades to diagnostics, actuators, and control systems would be required. Existing facilities have several major advantages over ITER and other future burning plasma devices: greater tolerance for ELMs and disruptions, which is necessary for the "learning curve" in avoiding them; greater flexibility for modification of the facility, which allows testing of multiple control techniques; and more complete diagnostic sets, facilitating the physics understanding and benchmarking of predictive models.

In addition, the emerging generation of **superconducting tokamaks** (EAST, KSTAR, SST-1, JT-60SA) will be crucial for demonstrating stable operation, free of ELMs and disruptions, in ITER-relevant plasmas for very long pulses. The larger existing and near-future devices (JET and JT-60SA) are an essential element of extrapolation from present devices to ITER, and techniques for avoidance of ELMs and disruptions must be tested to the fullest possible extent in these as well as the medium-size devices.

Ultimately, a **burning plasma experiment** such as ITER is required to develop and demonstrate avoidance of transient events in a fusion environment. The thermal and magnetic energy, heat flux, and potential for runaway electron generation during disruptions all become greater as the device size and toroidal field increase. On the other hand, the presence of strong plasma self-heating reduces the influence of external control. These conditions cannot be fully simulated in any existing facility, and continuing development in ITER will be an essential element in this research Thrust.

The US is well positioned to make a major contribution to this area of research with existing facilities. However, to address a meaningful fraction of these elements on the time scale needed for ITER will require a large commitment of resources, including:

- Significant increase in operating time for:
  - Developing and testing individual elements of instability avoidance and control.
  - Demonstration of integrated stability control under a wide range of conditions.
- Significant increase in staffing for:
  - Analysis and modeling for prediction of disruptions and ELMs.
  - Modeling and experimental tests of avoidance and mitigation strategies.
  - Validation of models suitable for extrapolation of these strategies to ITER.
  - Testing of systems to predict, avoid and mitigate disruptions on ITER.
- Modest facility upgrades:
  - Diagnostics, e.g., for runaway electrons and 3-D field effects.
  - Auxiliary systems for avoidance and mitigation, e.g., gyrotrons for localized current drive, mass injection systems for disruption mitigation, non-axisymmetric coils with improved spectrum flexibility.
  - Digital control systems, e.g., for more diagnostic inputs and greater real-time computing power.

Readiness

Disruption characterization can take advantage of a large existing database in present facilities. Further progress can be made in the near future, making use of non-linear 3-D MHD modeling codes and new diagnostics for high-speed imaging, halo currents, and runaway electron detection.

Disruption prediction in empirical forms for detection of growing tearing modes is employed at many facilities. Magnetohydrodynamic spectroscopy has been successfully used in off-line analysis of specialized experiments, and development toward its routine use in real time is needed. Many ideal and resistive MHD stability codes exist, and efforts are needed to adapt them for real-time analysis. Disruption prediction on current devices has not achieved the accuracy needed for ITER or DEMO.

Disruption avoidance is also in use at many facilities, typically in the form of a soft shutdown or retreat from high performance in case of loss of the desired operating state or onset of a tearing mode. Feedback-controlled stabilization of neoclassical tearing modes and resistive wall modes and feedback-controlled correction of error fields have been successfully demonstrated, but are not yet in routine use. Control strategies to take appropriate action based on calculated or measured approach to stability limits also are needed. Techniques to locally modify the pressure and current profile to avoid disruptions with small auxiliary power requirements have yet to be established.

Disruption mitigation by means of gas or pellet injection has received significant attention as a research topic, but is not routinely used in existing facilities. Techniques to suppress runaway electrons during a disruption have not yet been assured, but several approaches are under investigation. Decision processes that determine when and how to initiate a soft shutdown or a rapid shutdown — reliably but without a high rate of "false positives" — need to be developed.

Several approaches to **ELM control** are presently under investigation. The use of Resonant Magnetic Perturbation (RMP) coils for ELM control was pioneered by the US and has now been accepted as part of the ITER design. Work is also in progress to investigate other approaches such as pellet pacing and operating regimes without ELMs. However, the physics of ELM suppression is not yet well understood, and much additional research is needed to assure its extrapolation to ITER. A research thrust on control of transient events is well suited to the US fusion program. Key features of the existing US facilities include:

Actuators for equilibrium profile control, including heating, current drive, and torque from neutral beams, and several types of radiofrequency systems for heating and current drive.

- Versatile sets of non-axisymmetric coils (internal and external).
- Other actuators for direct suppression of instabilities, including localized current drive.
- Gas injectors and pellet injectors for rapid shutdown.
- Extensive and mature diagnostic systems for equilibrium and stability measurements, model validation, and real-time control.
- Sophisticated, extensible digital control systems.

Integration of research elements

The multiple research elements associated with prediction, avoidance, and mitigation of disruptions form part of a single effort. Off-line predictive modeling of stability limits and of disruption dynamics provide the theoretical foundation for this effort. Diagnostic measurements and analysis are necessary for scientific understanding as well as prediction of disruptions and detection of instabilities. Disruption avoidance requires hardware systems to modify the plasma equilibrium and stability, and mitigation requires systems for rapid delivery of large mass to the plasma core. Model-based control strategies are needed to use these tools for optimal results. These elements can be developed individually, but ultimately they must be integrated into a unified approach to disruption avoidance and mitigation for ITER and future burning plasmas.

Edge localized mode control, because of its critical importance to the first-wall lifetime, calls for a broad portfolio of approaches in the near term: active pacing by pellet injection, suppression by 3-D magnetic fields, avoidance by choice of operating regime, etc. Development of a theoretical understanding of the underlying transport and stability physics will provide the basis for assessment and projection of these approaches. Real-time assessment of plasma stability may ultimately be useful as input to ELM control techniques. Eventually these elements must be focused and integrated into a reliable ELM control system for ITER.

Connections to Other Thrusts

This research Thrust on control of transient events has strong connections to several others related to scientific understanding and control of plasma stability. A principal connection is to Thrust 5, since controlling and sustaining the desired operating state presupposes avoidance of disruptions. Control of equilibrium profiles, feedback control of instabilities, and scenarios for controlled shutdown of the discharge are dealt with in Thrust 5. Improved scientific understanding of the boundary layer plasma (Thrust 9) will be crucial in developing techniques for control or avoidance of ELMs. Accurate prediction of disruptions and ELMs requires accurate real-time diagnostic measurements and may even drive requirements for some diagnostics (Thrust 1).

The characteristics of the plasma facing components and their interaction with the plasma (Thrusts 10 and 12) are closely linked to the requirements for avoidance or mitigation of ELMs and disruptions. A device to test DEMO-relevant plasma boundary conditions (Thrust 12) could also serve to test control of transient events in DEMO-relevant conditions. Liquid walls or other innovations (Thrust 11) may ease the limits on peak heat flux. On the other hand, a thin-walled blanket for tritium breeding (Thrust 13) may have less capability to withstand the electromagnetic loads of a disruption. Use of liquid metal coolants (Thrust 13) may also limit the response time of magnetic sensors and stabilization coils.

The development of 3-D magnetic configurations such as stellarators (Thrust 17) represents a possible long-term approach to eliminating the disruption issue.

Thrust 3: Understand the role of alpha particles in burning plasmas

Fusion-produced alpha particles will constitute the dominant heating source in burning plasmas and thus open a new regime of investigation compared to all previous experiments, which are based on the use of externally applied heat sources. This significant change in plasma heating raises a number of issues unique to burning plasmas. Therefore, a detailed understanding of alpha physics is essential for extrapolations from ITER to a DEMO device and the employment of fusion as a future energy source.

Key Issues:

- Interaction with Background Plasma: *Will the alpha population in ITER cause significant plasma instabilities? If so, can these be avoided or tolerated? How will the alpha population interact with thermal plasma instabilities and turbulent transport?*

- Impact on Achieving and Sustaining Burning Plasma Operation: *How will alpha particles affect other heating and current drive methods? What additional heat load on the first wall will high-energy alpha particles cause? Will self-regulated steady-state regimes exist with strong alpha self-heating?*

- Measurement: *Can alpha particle phenomena and instabilities be adequately diagnosed in burning plasmas?*

- Control: *Can the alpha power be controlled for optimization of plasma profiles, currents, and flows? Can control techniques be developed to channel alpha energy directly to ion heating?*

Proposed Actions:

- Develop experimental scenarios to expand access to energetic particle behavior at high fast-particle pressure; explore energetic particle dynamics and interaction with instabilities, losses, and current drive.

- Identify operational regimes in burning plasma devices that are stable to alpha-driven instabilities and determine if alpha transport is tolerable in unstable regimes.

- Predict the alpha heating profile, alpha-driven currents, and impact on current drive requirements. Evaluate parasitic absorption of current drive power by alpha populations. Assess coupling of alphas with other fast ion beam components and the core plasma.

- Incorporate experimentally validated alpha physics transport models into integrated plasma simulation tools for the entire plasma.

- Simulate and validate alpha particle losses and their impact on first-wall integrity.

- Develop innovative high-resolution spatio-temporal measurements of the alpha particle energy distribution and instability mode structure.

- Exploit low-level Alfvén wave excitation as a spectroscopic diagnostic tool.

- Control alpha heating feedback loops for steady-state operation. Develop control techniques for alpha-driven instabilities and attempt to expand stable operating regimes. Facilitate helium ash removal. Attempt the direct transfer of alpha particle energy to the core fuel ions.

Introduction

This Thrust has three interrelated components. First, simulation and theory need to be improved and validated so that stable and unstable burning plasma regimes can be delineated and levels of alpha transport in unstable regimes predicted. Second, new advanced diagnostics will need to be developed for energetic particle populations and the instabilities driven by them. Third, based on the understanding gained in the previous two elements, alpha physics control techniques need to be developed for optimized fusion performance in ITER and DEMO.

Improvements in Alpha Physics Modeling and Simulation

The modeling and simulation components of this Thrust are essential to maximize the benefit of our ITER participation and to bridge the gap between ITER and DEMO. This effort is also highly synergistic with the diagnostic development component of this Thrust, since improved simulation results motivate better diagnostics and vice versa. Regular testing and validation of these simulation efforts against experimental data from new diagnostics is essential for the success of this effort. The first step is to develop realistic linear stability threshold predictions that can apply to alpha-driven Alfvén instabilities in the ITER regime. Since these modes always experience finite levels of damping, the reliable prediction of instability depends on comprehensive models of both damping and drive. Existing calculations require improvements in areas such as: sound-wave coupling, flow shear effects, better resolution of edge profiles and damping, improved radiative and continuum damping, inclusion of 3-D effects near the boundary, coupling to core micro-turbulence, and finite orbit-width effects. Accurate treatment of these effects becomes especially demanding for the small normalized fast ion gyroradius regime of ITER; in this regime, the most unstable mode number shifts to higher toroidal mode number ($n \sim 20$), and the number of modes available for destabilization at high n increases as n^2. This aspect significantly increases the resolution requirements and the complexity of linear stability threshold evaluation in ITER as compared to current experiments. Improved stability threshold prediction will be beneficial for ITER operational scenario development activities, integrated simulation efforts, and alpha physics control strategies.

In parallel with this effort, better nonlinear Alfvén stability models must be developed so that alpha transport can be assessed in regimes that are above the instability threshold. Such models will need to be upgraded to include multi-physics, multi-scale coupling to other plasma modes (e.g., sawteeth, microturbulence, resistive wall mode), and nonlinear avalanche dynamics. For large-scale plasma magnetohydrodynamic (MHD) instabilities (e.g., sawteeth and resistive wall mode), there are several concerns. One issue is that these modes can stimulate energetic particle avalanches and enhanced transient levels of alpha particle transport. A second issue is that energetic particles can temporarily stabilize core plasma MHD modes, allowing pressure and current gradients in the plasma to reach levels that can suddenly become unsustainable as energetic particle instability-driven transport removes the stabilizing influences of the alpha particles. This can lead to strong relaxation oscillations and prevent steady-state operation. Coupling between alpha-driven Alfvén modes and core plasma microturbulence becomes especially important for the high-n regime of ITER; the large collection of interacting Alfvén modes will drive nonlinear cross-scale couplings from meso to micro-scale dynamics. This feature will become more dominant in ITER than in current experiments, which predominantly observe moderate-n versions of these instabilities. This regime requires a paradigm shift to nonlinear gyrokinetic models that include all relevant wave-particle resonances and finite Larmor radius effects for both the thermal and energetic particle species. In addition, avalanche models need further development; the presence of many closely spaced modes could allow the energetic particle pressure gradient to progressively relax (flatten) from the center to the edge, implying rapid fast ion transport. To realistically model alpha transport in the edge plasma and divertor regions, improved equilibrium magnetic field models will be required that include the 3-D effects of toroidal field ripple, ferromagnetic materials, and coils to control resistive wall modes and edge localized modes; also, the effects of parasitic absorption of radiofrequency current drive power on alpha confinement will need to be assessed. The goals of such improved nonlinear models are to identify regimes that might be dangerous to the integrity of the plasma facing components, to predict the alpha heating profile evolution, and to evaluate the alpha particle influence on the driven current and its distribution. Ultimately, for ITER, with a fixed energy distribution (slowing-down, isotropic) for the alpha particles, it would also be expected that alpha transport scaling laws could be developed from simulations and experimental measurements. These could be useful for integrated modeling and for operational scenario selection.

The best simulation methods are in the process of being determined. Presently several approaches are in development. They include: (1) fully kinetic 3-D simulations that include the microinstability drive acting on the background plasma; (2) detailed 3-D hybrid MHD fluid simulations for the energetic particles and core plasma with the effects from microinstability treated in viscous and heat transport terms; and (3) the development of quasi-linear codes that can couple results from large fluid-particle codes to determine the transport coefficients that apply in the reduced 1-D profile evolution for the entire system. To achieve a deeper comprehension of the consequences of Alfvénic activity driven by alpha particles, reduced nonlinear theory will be incorporated in quasi-linear codes and then compared with the results of the larger simulations. The development path and viability for any of the above approaches will depend on regular validation against experimental data provided by the new diagnostics discussed in the following section. While validation tests in fully consistent burning plasma regimes (low ρ/a, $v_\alpha/v_{\text{Alfvén}} \geq 1$, central $\beta_\alpha(0) \approx 1\ \%$) may have to wait for ITER, there are many other aspects that can be checked from ongoing experiments.

Advanced Diagnostics for Alpha Physics

For the above improvements in simulation to have realism, an aggressive program of advanced energetic particle diagnostic development is needed. The goal of this effort should be to provide validation at appropriately resolved spatial and temporal scales and to stimulate improvements in modeling as unexpected phenomena are uncovered. The requirements for such diagnostics is an area with important overlaps to Thrust 1 and that much of the underlying development activity may occur under Thrust 1.

The first element of such a program is the measurement of the fluctuating electric and magnetic field instability mode structures. A number of techniques are now available for this task, such as heavy ion beam probes (already in use in the LHD stellarator experiment in Japan), Faraday rotation, polarimetry, motional Stark effect, and high-resolution charge-exchange recombination. However, these will require substantial commitments of resources for successful deployment. Such diagnostics would have cross-cutting use since they would also be applicable to studies of core turbulence and transport. Measurements of the instability mode structures are a critical step toward accurate prediction of alpha particle confinement. A second element of the advanced diagnostics effort is the high-fidelity measurement of the fast ion profile and temporal resolution of intermittent transport events. This information is necessary to understand the detailed power flows of fast ions, predict losses to the wall, and infer the alpha particle heating profile. A third element is the improved measurement of escaping fast ions. This involves covering more of the first wall and plasma facing components with energetic ion loss detectors so that greater spatial resolution of the heat loads is possible. Such information will be critical in predicting localized hot spots and in validating simulations of energetic particle losses. These three diagnostic efforts are expected to originate on existing non-ignited experiments, which will provide useful developmental test beds during the period (~ 15 years) leading up to D-T operation of ITER. The parallel development of neutron-hardened diagnostics for ITER will be necessary. These may be more limited in capability than what is possible on non-ignited devices; a greater reliance on simulation and synthetic diagnostic reconstruction will likely be required. Alpha particle-driven instabilities may also be useful in high-neutron-fluence fusion devices to infer q-profile information (via MHD spectroscopy) from the narrow, easily identified spectral lines of such fluctuations.

Experimental Studies and Collaboration

Access to experimental facilities is essential for the diagnostic and simulation developments that are mentioned above. In addition to US experiments (DIII-D, C-MOD, and NSTX), there exist international collaborations with JET and JT-60U, and future collaborations are anticipated with emerging devices such as KSTAR and EAST. Also, the capabilities developed in this Thrust are expected to be well suited to future D-D and D-T facilities that might be constructed as part of the US fusion program. The non-tokamak concepts (e.g., stellarator, reversed field pinch, field-reversed configuration) will provide further useful test beds for Alfvénic instabilities and energetic ion physics.

As new devices and plasma regimes become available, the detailed validation of ideal MHD predictions (n-number, radial structure, mode frequency), which has been successful on existing devices, will continue. Recent diagnostic advances in electron cyclotron emission, beam emission spectroscopy, and interferometry will allow the mode structure to be measured in greater detail. The future diagnostic challenge is to measure the perturbed currents and electric fields associated with these modes. The experimental challenge is to validate theory predictions of the internal structure over a wider range of parameters.

Quantitative understanding of the fast ion drive is improving; however, more systematic studies with greater variation of parameters will be required. Recent advances in fast ion D-alpha, Thompson scattering, neutral particle analyzers, and scintillator detectors are providing more qualitative information on profiles and fast ion losses. The diagnostic challenge for fast ion drive measurements is to provide more quantitative information on fast ion profiles and velocity distributions. The experimental challenge is to more directly calculate the instability drive and to compare this to the measured mode spectra.

Improved measurements of Alfvén mode damping are needed, especially continuum and radiative damping, coupled with validation. Alfvén mode excitation by external antennas and beat-wave sources in the ion cyclotron range of frequencies are promising techniques for damping studies. Recent rapid progress in internal mode structure measurements should enable better quantitative understanding. The diagnostic challenges in this area are to measure the short-scale structures that are predicted by theory and to identify the spatial structure and dissipation of medium-n modes. The experimental challenge is to validate the measured unstable and stable mode structure and compare the damping with calculated damping rates.

With respect to the nonlinear regime of Alfvén instabilities, the theory for their behavior near threshold has been developed. This theory requires extensive verification and validation; a variety of nonlinear phenomena are observed and predicted, including chirping, bursting, avalanches, and frequency splitting. The diagnostic challenge in this area is to measure phase space structures and fast ion transport or loss. The experimental challenge is to explore the threshold requirement for mode overlap and find methods for preventing its occurrence.

Controls for Alpha Physics

Successful validation of simulation tools with advanced diagnostics will form the knowledge base for the final and more exploratory aspect of this Thrust: the direct control of alpha particle effects to optimize fusion power performance. Control of the alpha heating source in ITER and DEMO will have high leverage in influencing plasma stability and behavior. Heating source controls are used extensively in existing devices to access improved confinement regimes, drive currents, stabilize MHD modes, and optimize plasma transport. However, unlike the heating power in existing experiments, alpha heating is self-generated and not as easily amenable to external control. New techniques will have to be developed and tested to accomplish such control. Experiments on DIII-D have indicated that focused electron cyclotron resonance heating can control fast ion-driven fluctuation levels. Beat-wave generation, with two ion cyclotron heating sources whose frequencies are closely spaced, has also been suggested as a way to deposit power in the range of Alfvén frequency fluctuations and modify alpha profiles. Control of the ion density profile (e.g., with pellets), *q*-profile, magnetic shear, and flow shear can be expected to have an impact on alpha-driven fluctuations. Enhanced loss via externally driven stochastic resonances (phase-space engineering) has been suggested for alpha ash removal and burn control. Damping of alpha-driven instabilities on other fast ion populations (created by neutral beam injection or ion cyclotron resonance heating) could be used for suppression. For example, injection of medium-energy positive-ion neutral beams near the ITER edge region could enhance Landau damping of alpha-driven instabilities and drive plasma rotation. These methods will need to be thoroughly assessed with simulation tools and studies in existing experiments prior to D-T operation in ITER.

A second area for alpha control studies, referred to as alpha channeling, involves methods for directly influencing the alpha particle heating feedback loops. The classical collisional slowing-down process for fusion-born alpha particles results in most of the alpha energy being transferred first to electrons, leading to electron temperatures that exceed ion temperatures. Fusion reactivity could significantly be improved if hot-ion regimes ($T_{ion} > T_{electron}$) could be sustained. Ideally, the goal of alpha channeling is to transfer the alpha particle birth energy to an appropriate set of waves, which could then directly heat fuel ions and simultaneously cause ejection of low-energy alpha particles (ash removal). Mode-converted ion Bernstein waves, Alfvén waves, and acoustic waves have been suggested as possible avenues for this energy transfer. Another mechanism, related to channeling, is to generate, in a controlled manner, phase-space "buckets" that spontaneously sweep in frequency as they transfer energy from the alphas to the background plasma. While the specific mechanisms and required wave spectra to facilitate alpha channeling remain somewhat speculative, the potential benefits motivate the inclusion of this area in the control component of this Thrust. As simulation tools and diagnostic methods improve, it is expected that more specific alpha channeling approaches can be identified and tested.

Conclusion

The achievement of dominant self-heating by fusion-produced alpha particles is a primary goal of the upcoming ITER and eventual DEMO projects. It is essential to monitor by active or passive measurements the properties of the alpha particle distribution function. Moderate Alfvénic instability can be tolerated and can, by itself, aid in the determination of the alpha particle distribution function. The redistribution of alpha particles within the plasma is tolerable, but significant loss of alphas is not. Methods of external control of the alpha particle build-up need to be developed, and the beneficial exploitation of the tendency of alpha particles to spontaneously transfer wave energy directly to the plasma should be investigated. Numerical transport and simulation tools, validated with results from present-day experiments, will be crucial for burning plasma experiments to predict the desirable range of operation and to assess the extent to which burn control techniques are feasible.

This ReNeW Thrust has overlap and close connections to a number of other thrusts. The development of new energetic particle and neutron-hardened diagnostics for the burning plasma environment is central to Thrust 1 and much of the diagnostic development described here will be done jointly between Thrusts 1 and 3; results from Thrust 1 will be critical to understanding alpha physics issues in future D-T plasmas. Thrust 5 will develop methods to control and sustain fusion plasmas; control techniques specifically for alpha physics should factor significantly into Thrust 5. Improvements in theory, simulation, and diagnostics for alpha-driven instabilities and transport should be coupled into Thrust 6, which has the goal of making overall advances in theory, simulation and measurements. Progress in alpha transport simulations will also be of use for Thrust 8, which addresses the integrated dynamics of burning plasmas. Alpha particle loss rates to plasma facing components are relevant to Thrusts 9–11, which cover various aspects of the scrape-off layer and plasma-material interfaces. Finally, although the primary focus of the present Thrust is ITER, the methods developed for simulating and measuring alpha-driven instabilities will have application to energetic ion phenomena in reduced-aspect-ratio tokamaks (Thrust 16), stellarators (Thrust 17), and configurations with minimally applied magnetic fields (Thrust 18).

Thrust 4: Qualify operational scenarios and the supporting physics basis for ITER

To attain high-performance states in ITER, we must form, heat, control, and safely shut down high-temperature plasma in a predictable and reproducible manner. Advancing our understanding of how to achieve these steps requires an integrated experimental campaign in plasma conditions similar to ITER, e.g., high-confinement mode (H-mode) plasmas with low input torque, equilibrated ion and electron temperatures, and low collisionality. Ensuring that ITER will efficiently achieve its objectives entails a high level of support by the US tokamak program. This requires upgrades to the tools for heating and current drive, particle control, and heat flux mitigation on existing tokamaks and, possibly, a new tokamak facility.

Key Issues:

- **Plasma initiation:** *What wall cleaning methods are suitable for ITER? Can techniques be developed to remove tritium from the walls without major loss of operating time?*
- **Transient phases:** *What is the energy and particle transport during current ramp-up and ramp-down, and how does it depend on the evolving current profile?*
- **H-mode access:** *What is the physical basis for extrapolating H-mode power thresholds adequately for ITER in the initial hydrogen and helium and eventual nuclear phases?*
- **H-mode sustainment:** *What is the physical basis for extrapolating H-mode confinement to plasmas with dominant electron heating, equal ion and electron temperatures, low collisionality, and low torque injection?*
- **Heating:** *Will the 20 MW of ion cyclotron resonance heating planned for ITER be effective, especially in transient phases and with plasma-antenna interactions taken into account?*
- **Fueling:** *What particle transport, pellet ablation, and fuel deposition models are appropriate to predict ITER density profiles and performance?*
- **H-mode pedestals:** *What physical processes form the edge pressure pedestal in H-mode plasmas, and how do they interact with core heat and momentum transport?*
- **Pulse length extension:** *What is the physics basis of the hybrid scenario, and does it extrapolate favorably to ITER? What is the optimum path to a steady-state advanced tokamak mode of operation, and what tools are needed to access it?*

Proposed Actions:

- Develop wall-cleaning techniques that are compatible with large, stationary magnetic fields.
- Measure and characterize transport physics during transient tokamak phases as well as in steady H-mode plasmas with plasma conditions similar to ITER.

- Determine minimum heating power required to obtain and maintain (1) H-mode during ramp-up and ramp-down, (2) steady H-mode with (small, rapid) Type III edge localized modes (ELMs) and (3) steady H-mode with good confinement (compared to scaling relations).

- Elucidate the physics and minimize interactions of ion cyclotron resonance heating antennas with edge plasmas.

- Develop models for particle transport and for gas and pellet fueling applicable to ITER.

- Test models of H-mode pedestal structure against experiment, determine the effect of pellet fueling and low flow/torque, and pursue coupling to core models.

- Understand the physics of the hybrid scenario to make reliable predictions for ITER, develop predictive understanding of steady-state modes of operation, and define requirements for implementation in ITER.

Success in ITER is predicated on the development of operational scenarios that achieve the necessary levels of plasma performance. Accessing high-performance states in ITER requires successfully forming, heating, controlling, and safely shutting down a high-temperature plasma in a predictable and reproducible manner. Achieving this requires greater understanding, gained through an integrated experimental campaign under plasma conditions as similar as possible to those expected on ITER, e.g., H-mode plasmas with low input torque, equal ion and electron temperatures, and low collisionality. *Uncertainties in the projected performance of ITER arise from many considerations, and the issues deemed most critical, and approaches for resolving or mitigating them, are described in the following sections.* With upgrades to the tools for heating and current drive, particle control, and heat flux mitigation on existing tokamaks, and a possible new tokamak facility, the US fusion community would be well-positioned to address these burning plasma issues along with others that may arise prior to or during ITER operation.

Wall preparation and cleaning. It is well known that the chemical composition of the plasma facing surfaces has a strong impact on the quality of tokamak discharges. Questions in this area are: *What cleaning methods can be used for ITER? Will wall-conditioning methods applied during and between discharges be effective?* A related question is: *Can techniques be developed to remove tritium from the walls, thereby extending the number of discharges before the operational limit on in-vessel tritium is reached?*

The number of wall preparation techniques available for ITER is limited due to its large steady-state magnetic field. One approach would be to use high-power radiofrequency waves near the ion cyclotron frequency. This technique has been used on a number of existing tokamaks, including the superconducting devices Tore Supra and EAST. A possibly more attractive approach would be to use electron cyclotron discharge cleaning (ECDC). In this method, millimeter waves are injected into the chamber vessel at or near the electron cyclotron frequency corresponding to the static magnetic field. In both cases, the resulting low-temperature plasma is effective at removing lightly bound elements from the chamber walls. The ECDC plasma tends to be most intense and effective at cleaning over only a relatively small region of the vessel at fixed frequency and magnetic field, so one of these must be varied. A thrust element addressing this issue would be to develop variable frequency gyrotrons. Thus, instead of slowly ramping the magnetic field strength to move the interaction region, which is the technique used on Alcator C-Mod, the frequency of the ECDC could be varied at fixed magnetic field (which ITER requires). Additional control can be implemented by application of a variable poloidal field. A possible ancillary benefit could be its use in freeing tritium from the chamber walls, which would be useful in controlling the tritium inventory in the vessel. Another benefit would be to resume research and development in the US on high-power, high-frequency, steady-state gyrotrons, an important enabling technology with additional uses in fusion experiments.

Transport during transient phases. Simulations of ITER scenarios have revealed the need for careful initiation and termination of plasma discharges. For the current ramp-up and ramp-down phases, simulations have been developed to model the evolution of the current profile, which is important for assessing magnetohydrodynamic (MHD) stability, but the results depend on electron thermal transport during these transient states, which is poorly understood. The issue is: *What is the energy and particle transport during ramp-up and ramp-down phases, and how does it depend on the evolving current profile?* Uncertainty concerning transport during current ramp-up and ramp-down can be reduced by coordinated research on this poorly studied topic. To be most relevant to ITER, it is desirable to use dominant electron heating, likely from radiofrequency sources.

Sawtooth mixing and temperature recovery between sawtooth crashes constitute a transient condition that may be of more importance for ITER than present-day devices. To study transport within the sawtooth mixing region, high time resolution is required for measurements of magnetic pitch angle, ion temperature, and flow speed. Experiments need to vary the heating mechanism between thermal (e.g., electron cyclotron resonance heating) and fast ion thermalization to validate models of the effect of suprathermal sawtooth stabilization. Like sawteeth, ELMs will effectively produce significant radial transport in the periphery, and the influence of ELMs on the plasma boundary needs to be quantified for ITER.

H-mode access and dependence on ion species. Access to the H-mode is essential for ITER to fulfill any part of its experimental mission. An example of a low- to high-confinement (L-H) transition, including the observation of ELMs during the H-mode phase, is shown in Figure 1. A high-priority research activity is determining the heating power required for attaining various regimes: (1) the transition threshold from L-mode to H-mode and from H-mode to L-mode, (2) steady H-mode plasmas with (small, rapid) Type III ELMs, (3) steady H-mode plasmas with "good" confinement, as determined by standard scaling relations, and (4) H-mode access and back transition during the plasma current ramp-up and ramp-down phases. These studies need to be done in ITER-relevant plasma conditions. The isotope mass and species scaling (i.e., hydrogen and helium plasmas) for the heating power required to achieve the regimes listed are also of high value when considering the nonnuclear phase of ITER.

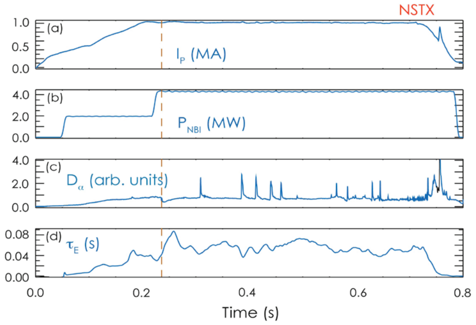

Figure 1. Example of transition (dashed line) from low to high-confinement mode on NSTX: (a) plasma current, (b) neutral beam injection power, (c) edge D_α recycling light with ELMs indicated by spikes, and (d) energy confinement time. (Figure courtesy of Stan Kaye.)

Research should focus on developing a physical basis for extrapolating H-mode power thresholds reliably and accurately and should develop strategies for minimizing the power requirements. Examples of the latter include pellet injection, plasma shape modification, inboard gas puffing, and varying the plasma current ramp-rate. Progress will require improvements in edge diagnostics, particularly those that measure profiles of relevant quantities such as density, temperature, and flows. Characterization of edge fluctuations leading up to L-H transitions is also important. Ample heating power should be available in multiple forms (neutral beams, radiofrequency waves, etc.). Efforts should be made to resolve differences in power threshold results that may arise among different heating schemes.

H-mode sustainment. Plasma conditions in ITER — dominant electron heating, electron-ion temperature equilibration, low neoclassical collisionality and low torque injection — will differ from those in most present-day tokamaks. The issue is: *What is the physical basis for extrapolating H-mode confinement to this reactor-relevant condition?* The definition of reactor-relevant conditions should include those processes necessary to reduce the heat flux on plasma facing surfaces, such as the suppression of edge localized modes and radiative divertor operation.

Since plasmas are complex physical systems, determining the scaling of transport phenomena with dimensionless parameters is a valuable tool. A transport extrapolation to smaller relative gyroradius (ρ^*), and perhaps smaller collisionality (ν^*), while keeping the other dimensionless variables fixed is the preferred scaling path. A high priority for future transport experiments is to use dominant radiofrequency heating to better simulate the burning plasma regime with strong electron heating, equal ion and electron temperatures, and low torque injection.

Rotation and velocity shear have a beneficial effect on confinement and play a role in the L-H transition and the stability of various MHD modes. While extrapolation of intrinsic rotation from an inter-machine database appears favorable, it is important to determine the origin of spontaneous rotation and validate its size scaling. Recent observations of flow drive from mode conversion of radiofrequency waves look promising, and electron cyclotron waves and lower hybrid waves should be pursued as potential profile control tools. Experiments also need to validate models of the effects of resonant and nonresonant drag from non-axisymmetric magnetic fields. The counter-current offset to rotation from neoclassical toroidal viscosity could potentially cancel out externally driven rotation in the co-current direction and needs to be documented for extrapolatation to future devices.

Heating and fueling. During the ramp-up and flattop phases in ITER, heating and fueling must be carefully programmed to reach the high-gain state where alpha particle heating becomes dominant. ITER will be heated by a combination of neutral beam injection and various radiofrequency heating methods (ion cyclotron, electron cyclotron and possibly lower hybrid waves). The heating physics of these methods is well understood, and the full injected power is expected to be effectively absorbed by the plasma; however, there are plasma interface or technological issues associated with each of the methods. Of particular interest in the US is the interaction of the ion cyclotron antenna with the edge plasma. The questions are: *What will be the nature of the radiofrequency sheaths formed by the ion cyclotron antenna? What will be its effect on impurity production and the effect of impurities on antenna operation? What will be the antenna impedance and will the required antenna voltage be acceptable?*

The specific Thrust element is a program focused on the interaction of ion cyclotron antennas with the edge plasma, including development, verification, and validation of models for radiofrequency sheaths and their effects. Closely coordinated work using experimental, theoretical, and computational tools is required. It is expected that this activity will point toward the need for advanced antenna designs, and these should be implemented and their performance validated in existing machines. It is also expected that methods of reducing the antenna voltage below breakdown levels will be developed and validated within the frame of this Thrust element.

Control of the density and impurities and, to the extent possible, their spatial profiles is important for the transient and steady discharge phases. The limits of gas fueling at high neutral opacity and the role of transport in setting the density profile need to be explored. The fact that the particle transport differs among species could be leveraged to isolate control of particular fueling and impurity species. Questions to be resolved in this domain are: *What is the particle transport, including impurity transport, for the various ITER operating scenarios, and what pellet ablation and fuel deposition models are appropriate? Can diagnostics be developed to determine the core deuterium and tritium ratio to aid burn control and as an adjunct to determining tritium retention*? The specific activity in this Thrust element involves measuring and characterizing transport in transient phases and validating models of fueling via (inside launch) pellet injection. Existing tokamaks properly outfitted with inside launch pellet fueling and an appropriate diagnostic set should suffice for carrying out this mission. In addition, advanced fueling techniques, e.g., compact tori injection, should be pursued.

H-mode pedestals. The ability of the ITER device to achieve its high fusion gain (Q=10) mission depends on having an edge pedestal sufficient to maintain high core confinement. Typical profiles for the density and temperature edge pedestals are shown in Figure 2. The questions are: *What is the physics of the edge pressure pedestal in H-mode plasmas, and how does the pedestal interact with core heat and momentum transport?* Recent modeling has reproduced the height of the H-mode pressure pedestal in several tokamaks by combining a prediction of edge MHD stability limits from peeling-ballooning mode theory with an empirical scaling for the pedestal width. Transport can play a role when ELMs are mitigated or suppressed so that the pressure gradient is held below the threshold for peeling-ballooning instability. Models of the H-mode pedestal structure and of the complete ELM cycle need to be further developed and thoroughly tested against experiment. This includes the effect of pellet fueling and low flow/torque. In addition, models of the H-mode pedestal structure need to be coupled to core models of heat and momentum transport to arrive at a comprehensive, predictive, and validated confinement model.

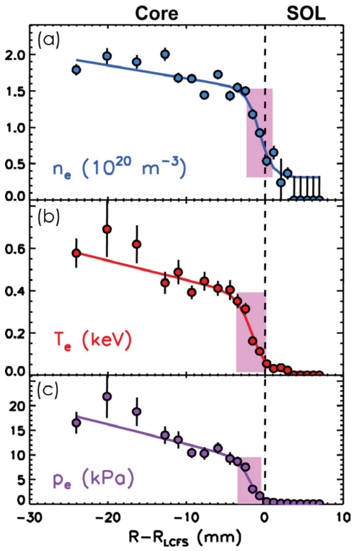

Figure 2. Example of an H-mode pedestal (shaded region) near the plasma boundary (dashed line) from Alcator C-Mod: (a) electron density, (b) electron temperature, and (c) electron thermal pressure. (Figure courtesy of Jerry Hughes.)

Other outstanding issues that need to be studied include the effect on the quality of the H-mode pedestal with (1) helium or hydrogen operation, (2) near-unity ratio of input power to H-mode threshold power, (3) relatively small separation between the primary and secondary separatrices, and (4) high opacity to edge neutrals, since ITER will not likely be dominated by edge fueling as are tokamaks today.

Scenarios for extended pulse length. Two operational scenarios are foreseen in ITER for extending the pulse length: the hybrid scenario and the advanced tokamak (AT) scenario. The hybrid mode is characterized by a broad current profile with a central safety factor ≥1. Improved stability and confinement occurs, allowing the current to be reduced below the value nominally needed for standard H-Mode without loss of performance. While promising for ITER, the physics of the hybrid scenario is not completely understood. The task is to: *Develop predictive physics understanding of the hybrid scenario, including whether current drive tools are needed to maintain a broad current profile with central safety factor ≥1 and to determine if performance extrapolates favorably to ITER-relevant regimes*. In particular, the role of MHD activity in sustaining the current profile in today's hybrid regimes needs to be better understood.

Steady-state AT modes for ITER require even further development. The preferred mode has nearly zero or slightly reversed magnetic shear, with the central safety factor ~ 2. As in the hybrid scenario, enhanced stability and confinement are key features with the additional benefit of a large bootstrap current fraction. Such AT modes have been obtained in existing tokamaks, but only for durations of a few resistive skin times (see Figure 3). The research required is *to develop a predictive physics understanding of steady-state AT scenarios so that they can be confidently extrapolated to ITER*. An important issue is the degree to which the duration for such AT regimes can be extended for times long compared with the resistive diffusion time.

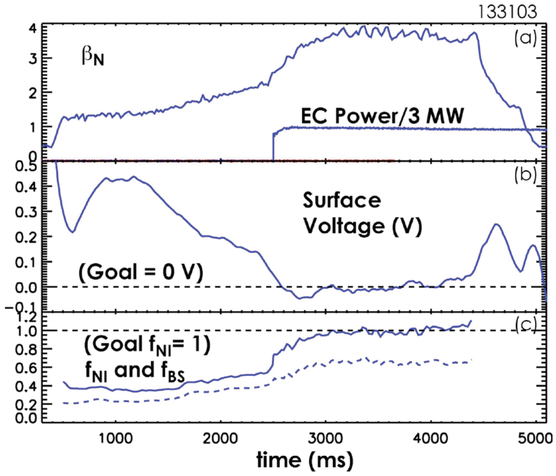

Figure 3. Example of a steady-state advanced tokamak discharge from DIII-D: (a) normalized beta and electron cyclotron current drive power, (b) surface loop voltage, and (c) noninductive and bootstrap current fractions. (Figure courtesy of Chris Holcomb.)

Developing steady-state scenarios for ITER is also in line with the research required for a tokamak DEMO, since DEMO operation is foreseen to be steady state. However, the fusion gain of 5 targeted for ITER's steady-state mode is well below that required for an economic power plant. Thus, the thrust toward steady-state research should be aimed not only to enable the ITER steady-state goal to be reached, but also to establish the basis for a steady-state DEMO. The difference between ITER's needs and those of a DEMO can be measured by the bootstrap current fraction (with the remainder of the plasma current driven by neutral beams or radiofrequency waves); in ITER the bootstrap current fraction is about 50%, whereas in DEMO a bootstrap current fraction of 80% or more is projected.

Much of the research required for extending the pulse length on ITER can be carried out on the existing domestic tokamaks with enhanced capabilities. Sufficient current drive tools should be added to ensure that the physics of both hybrid and AT scenarios could be adequately investigated. As a first step, this could be accomplished by doubling the electron cyclotron current drive power on DIII-D, the neutral beam current drive power on NSTX, and the lower hybrid current drive power on Alcator C-Mod. As different current drive methods have distinct advantages in their ability to tailor the current profile, future steps may require diversifying the mix of methods on each facility. Additional contributions will be made by the international programs, in particular, JET and the new superconducting Asian tokamaks with their much longer pulse length. However, to maximize the prospects for ITER's success, it is necessary as far as practical to develop *integrated* scenarios, i.e., scenarios that simultaneously achieve the high-performance core, edge, and steady-state conditions as measured by the standard dimensionless parameters (e.g., β, ν^*, and ρ^*). Achieving such integrated performance and the underlying scientific understanding may require a new US facility with lower ρ^* than is currently achievable.

Required facility capabilities. Answering these types of burning plasma questions prior to and during ITER operation will require supporting tokamak research facilities with the tools necessary to reproduce, study, and control plasma conditions as similar as possible to those in ITER (with the exception of tritium usage and fusion self-heating). Desired attributes for these supporting facilities include:

- Sufficient heating with little torque injection to approach the sustained H-mode conditions with dominant electron heating and dimensionless parameters anticipated for ITER.
- Sufficient mix of heating power to vary the effects of fast ions, thermal particles, and radiofrequency power on the H-mode pedestal, plasma rotation, and flow shear.
- Sufficient current drive power to access sustained safety factor profiles that approximate the main ITER operational scenarios and drive ≥ 50% of the total plasma current.
- Pulse length many times the resistive diffusion time.
- Extended capability to measure electron and ion-temperature fluctuations, and high spatial resolution profile diagnostics (e.g., improved current profile diagnostics) to support research in H-mode threshold and sustainment, including transient conditions.
- Enhanced ability to control particles at the plasma edge and fuel in the plasma core to resolve the remaining physics issues in heating and fueling for application to ITER.
- Methods to mitigate or suppress edge localized modes in low-collisionality plasmas.
- Radiative divertor operation.
- Ability to determine adequate methods for wall preparation and cleaning.

Addressing the critical issues for ITER identified in this Thrust requires a focused and well-coordinated research plan. To obtain maximum productivity from the domestic program in addressing ITER issues, the three US tokamaks should be upgraded to realize as many of the noted attributes as possible. Where appropriate, particularly in the area of pulse length extension, partnerships should be formed with new international tokamaks. On the longer term, it is essential to address the issues confronting ITER in a device with scenario capability that permits — and indeed requires — development of *integrated* solutions. The ability of the world tokamak program to provide the platform(s) for developing integrated solutions needs to be evaluated. If the capability of planned and existing tokamaks falls short of those needed for developing integrated solutions, construction of a new domestic facility should be undertaken. Such a facility could be compatible with the high-priority goals in other thrusts, e.g., Thrust 8.

Thrust 5: Expand the limits for controlling and sustaining fusion plasmas

A fusion reactor operating in a robust steady state at pressures beyond conventional stability limits would bring enormous advantages in terms of economy and efficiency. To achieve and maintain such conditions in a tokamak will require an advanced tokamak scenario with an unprecedented level of active control.

Key Issue:

> *What is the highest performance level of a tokamak fusion plasma that can be controlled and maintained for an unlimited period of time without unacceptable transients?*

To maximize performance and achieve the necessary level of control, significant advances will be required in three areas:

Sensors: Diagnostic techniques for measuring the plasma state to enable regulation that are capable of operating robustly in a sustained-duration burning plasma nuclear environment.

Actuators: Means to sustain and affect the plasma through flexible heating, current drive and fueling systems, thus enabling reliable burn control and profile regulation.

Algorithms: Mathematical schemes using control-level reduced models to translate measurements into actuator commands, thus providing robust and quantifiable assurance of sustained operation.

The control-specific engineering, physics, and mathematical solutions require a high degree of cross-community integration. The goal requires developing and integrating these three elements to enable operation in close proximity to or beyond passive stability limits. The integrated solutions must maintain robustness to transients and avoid (or mitigate when necessary) serious off-normal events. Although we will learn from present devices and ITER, these cannot operate in a regime requiring the active control level of a steady-state demonstration reactor.

Specific issues cover understanding and solutions for:

- Active control to robustly sustain the plasma in steady state, including burn and fueling control.
- Robust active stabilization of instabilities and transient fluctuations.
- Regulation of the power flow to material surfaces.
- Active prediction, avoidance, detection, and response to disruptions and fault events.

Proposed Actions:
Short-term: Begin development of nuclear-robust control-specific diagnostics, high-performance actuators, reduced models, and robust control algorithms with appropriate enhancement and exploitation of presently operating devices. Increase integration of relevant areas of expertise in diagnostics and actuators for control requirements, control model development, control algorithm mathematics, computational simulations, and physics understanding.

Medium-term: Use new experiments to demonstrate solutions with extended pulse duration in deuterium (D-D) plasmas, through international collaboration where possible.

Long-term: Use ITER and new deuterium-tritium (D-T) devices being proposed to develop and demonstrate the integrated control solutions required for DEMO.

The key element that distinguishes the advanced tokamak (AT) and related scenarios from more conventional limited-pulse tokamak operation is the unprecedented level of active control required for robust high-performance operation. The ultimate goal of a control system is to control the power flow in a reactor safely and efficiently, by controlling the plasma. The central aim of the Thrust then is to enhance, develop, and demonstrate the enabling science and technology for controlling and sustaining fusion plasmas in steady state and in close proximity to or beyond passive stability limits. This will require progressive development and integration of control solutions, diagnostics, and control actuators for a set of plasma parameters that are strongly interrelated: global parameters; plasma shape; current density profile; density and temperature profiles, rotation profile; D-T ratios; and impurity fractions. Control of each element requires technologies for:

(i) Diagnosis of the current state. Generally the diagnostic involves a set of measurements from which the parameters of interest can be derived.

(ii) An actuator to modify the state. Typically this involves controlling several kinds of input to the plasma that can be related to the parameter needing to be controlled.

(iii) An algorithm to translate the required change in the state to the actuators controlling the input to the plasma.

The necessary control will require a unified and comprehensive system in which the required sensors, algorithms, and actuators are fully coordinated to diagnose and modify the parameters, profiles, and their nonlinear interactions in a largely self-heated and self-driven tokamak plasma. The facility for robust detection, avoidance, and response to significant transients and off-normal events must be an integral component of the overall goal of maintaining steady state. This Thrust will combine expertise from different communities to produce the integrated control-specific engineering, physics, computational methods, simulations and mathematical solutions essential to the success of a robust and reliable control system. The vision would be akin to that of modern high-performance aerospace engineering systems that also operate beyond passive stability limits. The challenges in a fusion reactor are that the fusion environment is much more hostile to sensors and actuators, and the system is more fully self-driven.

In the short term, the Thrust will exploit presently operating devices — DIII-D, C-Mod, NSTX and other experiments — and increase integration of relevant areas of expertise in diagnostics and actuators, control model development, control algorithm mathematics, computational simulations, and scenario physics understanding. In the medium term, new superconducting tokamaks, both coming on line and being proposed, will be used to develop and demonstrate control solutions for extended pulse durations. It is expected that the tasks can be achieved either through collaboration with the new Asian superconducting facilities or in a possible domestic long pulse tokamak experiment. In the longer term, ITER will be used to develop and demonstrate the control solutions under fusion conditions. Finally, new experiments, in which the pressure and associated bootstrap current are produced largely from alpha heating, with at least a moderate, but preferably long or steady-state pulse length, are required to demonstrate the control solutions needed for a power producing reactor. Also required is a high fluence facility for testing survivability of diagnostics and actuators. Approaches for realizing these requirements are described in Thrust 8 and Thrust 13.

Specific issues cover understanding and solutions for:

Active steady-state control: *How near optimal can the plasma profiles and bulk parameters be robustly maintained in sustained steady state?*

Startup and shutdown: *Does a safe and reliable path exist from low current and low pressure to the required highly self-regulated, high-performance burning plasma state?*

Burn control and thermal stability of the operating point: *With what level of dynamic performance and flexibility can thermal stability be provided in a burning plasma?*

Robust active stabilization of instabilities and transient fluctuations: *How close to or how far beyond stability limits can ATs operate with maximum efficiency and negligible probability of control loss using robust active control?*

Regulation of the power flow distribution to material surfaces: *What level of power flow regulation can be achieved in the presence of plasma transients?*

Active prediction, avoidance, detection, and response to off-normal and fault events: *Can the probability or occurrence be reduced to levels required for a power plant, and reliable response algorithms be developed for acceptable device protection?*

The goals, challenges, and research plans for each of these are discussed in turn. Nevertheless, the issues are generally interrelated and need to be addressed by an integrated approach. Many of the diagnostics and actuators are used for different control tasks and thus require the same development; however, the specific goals of what is being optimized vary and these need to be balanced by an integrated control system. A few key examples are given in the sections following. More comprehensive research needs are described in the Theme 1 and 2 Chapters on measurements and auxiliary systems.

Demonstrate active control of the plasma equilibrium state maintained in steady state close to an optimum performance configuration: *How near optimal can the plasma profiles and bulk parameters be robustly and efficiently maintained in sustained steady state?*

The basic control challenge is to maintain the plasma parameters and profiles in a steady state optimized for a given desired confinement and stability, with sufficiently high efficiency to enable economic power production.

Specific Challenges:
A tokamak reactor based on highly optimized AT scenarios will ultimately require control of the equilibrium pressure, current, and rotation profiles in a high-fluence nuclear environment maintained in steady state. The control system sensors, algorithms, and actuators must be capable of responding on multiple time scales, from magnetohydrodynamic (MHD) (μsec and msec) scales, to current diffusion and transport (seconds) scales. The components must also survive for long time scales during which wall interaction, and atomic and nuclear chemistry effects, can modify the plasma characteristics. These will require real-time equilibrium reconstructions and sufficient diagnostics and actuators capable of measuring and modifying key equilibrium profiles: the plasma species density n_k and temperature T_k for each species k, and pressure $p = \Sigma n_k T_k$, current density j, and rotation $\mathbf{v}$ profiles. Rotation is of interest since it is known to affect stability and transport, and could potentially provide a powerful knob for affecting density and temperature profiles. Presently, the required actuators for controlling the full complement of these, either directly or indirectly, are either nonexistent or not sufficiently capable for DEMO-level plasmas. Density control will rely on fueling techniques, particle transport and the possible effect on particle transport of the current drive techniques and the alpha heating, which all require further development and increased understanding. The current must be sustained noninductively (i.e., without using the transformer) and its profile will largely be determined by the bootstrap current. The challenge for the actuators (whether radiofrequency systems or neutral beams) will be to deliver the supplementary current to the appropriate location with high efficiency. While the physics of current drive is well understood, the ability to place it where required is much more poorly demonstrated. Control of density and rotation profiles is the least well developed; radiofrequency techniques for rotation control are at a rudimentary level of understanding, as is the knowledge of the intrinsic rotation, especially in the presence of alpha heating. Tailoring the rotation profile appears feasible using a combination of techniques, but has not been demonstrated in a predictable way. Control of edge gradients in all quantities is expected to be essential, requiring diagnostics for the first-wall conditions, and in situ wall conditioning will be needed to control external conditions. Control system algorithms will need to be capable of optimizing and providing closed loop signals to radiofrequency systems, fueling systems, external coils, and other actuators from necessarily limited diagnostic data supplemented by timely simulations and control-level models.

Additional serious challenges arise in a burning plasma, where the pressure is largely determined by alpha particle heating and direct control of the profile may be impossible. Indirect control will depend on developing sufficient understanding of the coupled state to develop model-based controllers and algorithms to achieve the needed level of control.

Research Plan:

<u>Short-term:</u> Develop and test improved actuators. Examples include high-efficiency, steady-state, high-power, flexible (frequency variable) gyrotrons for effective electron cyclotron heating (ECH) and current drive (ECCD), with extension to higher density operation, and rotation control through lower hybrid (LH) and ion cyclotron range of frequency (ICRF) waves. Demonstrate full closed loop equilibrium control on existing experiments.

<u>Medium-term:</u> Develop new diagnostics and radiofrequency actuator systems scalable to high-fluence environments, for example, ICRF and LH systems that minimize interaction with the scrape-off layer plasma, and innovative ECCD steering methods. Demonstrate integrated control methods in steady-state D-D plasmas and test new actuators.

<u>Long-term:</u> Demonstrate fully functioning integrated control methods, and develop and test new actuators and diagnostics in ITER D-T plasmas. Extend to strongly alpha dominated, DEMO-prototypical environment.

Develop and demonstrate reliable procedures for robust startup to a largely self-heated and self-driven steady-state configuration and reliable shutdown protocols: *Does a safe and reliable path exist from low current and low beta to the required highly self-regulated, high-performance state envisaged in a burning plasma?*

Startup and shutdown of tokamak plasmas is prone to heightened possibility of disruption if not done carefully. Generally, the pressure, current, and plasma shape need to be ramped up to a largely self-sustaining bootstrap current and alpha heated state, and terminated when necessary by a careful ramp-down.

Specific Challenges:

While existing experiments routinely reach bootstrap dominated plasma current states using auxiliary heating methods, the alpha-heated state will provide unique challenges. Achieving the desired current profile with large bootstrap fraction will require optimization of the current drive tools (whether radiofrequency or neutral beam injection) to meet the current drive magnitude and location requirements within the available power constraints.

Research Plan:

<u>Short-term:</u> Explore more fully from current experimental databases and additional experiments the dependence of disruptivity on startup rates across multiple machines for developing control-level models. Continue bootstrap current startup experiments and other novel ohmic-free startup scenarios in existing experiments.

<u>Medium-term:</u> Test new control methods for startup to full noninductive current and high pressure in D-D plasmas.

<u>Long-term:</u> Test startup scenarios to full steady state in ITER D-T advanced scenarios with significant alpha heating and corresponding bootstrap current. Extend to alpha-dominated heating and bootstrap current.

Demonstrate burn control and control of the operating point thermal stability with sufficient flexibility to regulate power output to accommodate external demands: *With what level of dynamic performance and flexibility can thermal stability be provided in a high-performance burning plasma?*

A power producing reactor will need to operate at a constant total power output determined by the external load and economic considerations, but must be sufficiently flexible to vary the operating point as the external load changes. In addition, thermalized He "ash" needs to be removed to avoid quenching the reaction. Power control is achieved by controlling the isotopic mix, density profile, and ion-temperature-dependent fusion cross-section by active fueling and pressure profile modification. Thermally stable operating points exist in which a positive excursion in the ion temperature reduces the fusion reactivity, and the plasma control system must be capable of reaching and smoothly transitioning between these points. However, thermally stable operation is not necessarily the most efficient, and the option of active feedback control capable of maintaining the operating point at passively unstable values is desirable. Ash removal is a challenge and methods need to be developed for selective removal of cooled He nuclei. Techniques for deep fueling need to be developed, along with a deeper understanding of particle transport. Auxiliary heating schemes can modify particle transport, including impurity transport and density peaking. A predictive understanding of these observations, which would quantify the power required to achieve specific values, requires new research.

Specific Challenges:

Precise control of the total power output requires determination of the currently existing density and temperature profiles in a burning environment, and actuators for then reaching the desired operating point. Actuators are also required for maintaining passively unstable operation; even for passively stable operation, externally mandated variation in power output may require transition through an unstable region. Removal of alpha-particle ash requires new diagnostics for measuring its density profile and a mechanism for selectively removing the ash, ideally by increasing cross-field transport of non-hydrogenic ions without degrading overall confinement. Present ideas for achieving this, using various plasma instabilities, are not much beyond conceptual stages.

Research Plan:

Short-term: *Develop understanding of fuel deposition and particle transport in current experiments sufficient for predictive control of the profile and isotope mix through fuel injection. Develop advanced fueling techniques, such as pellets or Compact Toroid injection capable of deep fueling in reactor-grade plasmas. Develop control-level models based on simulations for identifying and controlling the operating point through profile control and fueling, and develop diagnostics and actuators for controlling alpha ash.*

Medium-term: *Test and further optimize in a long pulse D-D device the control models for maintaining and varying passively stable operating points, and test diagnostics for measuring the alpha particle ash profile and novel ash removal techniques.*

Long-term: *Test integrated system and control models for maintaining and varying passively stable and unstable operating points, ash diagnostics and removal techniques, and deep fueling in ITER. Extend to higher alpha heating, and high fluence D-T environments.*

Develop the means for and demonstrate robust active stabilization of instabilities and fluctuations: *How close to or how far beyond stability limits can ATs operate robustly with maximum efficiency and negligible probability of control loss using active control?*

While the requirement for active control of instabilities to enable increased fusion performance beyond the passive limits assumed in conventional scenarios greatly increases complexity, the potential performance gains can be large. A precedent exists in modern aerospace designs, which have similarly evolved from passively stable early aircraft with limited maneuverability to modern high-performance fighters that operate in an extended but unstable dynamic range, maintained by a complex control system.

Specific Challenges:
Operation in passively unstable regions requires a robust control system; loss of control can quickly have serious consequences. While this is already routine in the case of axisymmetric MHD stability, with the passive plasma elongation limits exceeded in most tokamaks but maintained by an active feedback system, operation beyond the passive beta limits is still an ongoing research area. Reliable detectors and actuators that can survive in a burning plasma are needed for the major beta limiting instabilities: resistive wall modes (RWMs), neoclassical tearing modes (NTMs), edge localized modes (ELMs), core "sawteeth," and fast ion driven Alfvén eigenmodes (AEs). The key actuators are envisaged to be radiofrequency waves and active non-axisymmetric field coils. The coils and especially the magnetic sensors need to be close to the plasma, and thus must be protected from the hostile, high-fluence nuclear environment. Radiofrequency systems for locally maintaining stability require precise localization. For ECH, tracking systems and mirrors are required and need to be shielded. While the AT scenarios envisaged generally require suppression of RWMs, NTMs, and AEs, sawteeth and edge instabilities may play a positive role in eliminating impurities, including alpha ash, and regulating the profiles in the core and edge respectively. This must be balanced with effects on core performance and power handling. Thrust 2 will develop solutions for ELMs on ITER, but extrapolation to reactor conditions will require further research. Sawteeth and ELMs also have strong nonlinear couplings with RWMs, NTMs and AEs, so their regulation must be integrated into the full active feedback control system.

Research Plan:
<u>Short-term:</u> Continue development and optimization of radiofrequency actuators and sensors for detecting early onset and control of fluctuations resulting from instabilities in existing experiments and initiate efforts into new methods that can scale to a reactor. For example, develop high-power high-efficiency gyrotrons and remote steering required for MHD stabilization, and test other methods for MHD stability control.

<u>Medium-term:</u> Test and optimize integrated control system options for controlling all the key instabilities in D-D plasmas for long pulses, progressively raising beta above the passive limits. This should be done in collaboration with new Asian tokamaks where appropriate.

<u>Long-term:</u> Test integrated control system options for controlling all the key instabilities in D-T plasmas for long pulses above the passive beta limits in ITER. Extend to alpha-dominated D-T plasmas above the passive beta limits. Tests in long pulses and high-fluence environment would be ideal.

Develop the means for and demonstrate regulation of the power flow distribution to material surfaces in the presence of core plasma transients, while keeping each component within acceptable tolerances: *What level of regulation of the power flow distribution can be achieved in the presence of fluctuations from plasma transient events?*

The power flow from a fusion reactor must be in a useable form and is limited by the capability of materials to handle the influx of radiation, neutrons, heat, and energetic charged particles. The spatial and temporal distribution of each of these components needs to be controlled sufficiently to avoid exceeding local material limits.

Specific Challenges:
The different elements comprising the power flow have different spatial distributions. Radiation and neutron fluxes are essentially volumetric sources and absorbed by the first wall and shielding. The heat flux from thermal particles is largely directed to specific target divertor plates by the plasma edge magnetic geometry. With current technology, the power flux allowable on divertor plates is ~10 MW/m^2 and is normally most limiting. Reliable control is needed to balance and maintain the various fluxes at manageable levels in the presence of temporal fluctuations from plasma instabilities — notably sawteeth and ELMs, but also from other sources. Presently envisaged solutions for ITER using radiative divertors, for example, may not be sufficient for DEMO. A search for solutions is a primary aim of Thrusts 9 and 12. Implementation of these solutions in a high-performance steady-state system is a major challenge for this Thrust.

As discussed earlier, the actuators for maintaining and controlling the equilibrium state indirectly control the power flow distribution. Control of the power flow requires sufficient confidence in understanding the dependence of transport on the equilibrium configuration and various fluctuations that control-level models of this dependence can be extracted. Presently there are serious gaps in the required understanding, especially in the edge region and in the presence of ELMs. Precise control of the magnetic field strike points at the divertor, and of detachment of the radiating divertor, is crucial. Diagnostics for the divertor region and for fast ion outflux require major development. The control system is needed to balance the requirements of maintaining the power flows within acceptable tolerances and minimizing the recirculating power required by the actuators. This necessitates developments in control algorithms coupled with understanding of the limitations and capabilities of sensors and actuators, as well as the interrelations between the plasma equilibrium, amplitudes and frequencies of transients, and the resulting power fluctuation levels.

Research Plan:
Short-term: Develop the control required to implement divertor solutions, obtained in Thrusts 9 and 12, in existing experiments that can scale to reactor conditions.

Medium-term: Test elements of the control system for regulating power flow in a steady-state D-D environment with divertor control and fluctuation detection and response.

<u>Long-term:</u> *Test the control system for regulating power flow in a D-T environment in ITER, with divertor control and fluctuation detection and response. Extend to demonstration of the complete integrated power flow regulation system, including control of fluctuations within the tolerable levels, in an environment dominated by alpha particle heating, and at the power flow levels expected in or near those in a commercial reactor.*

Develop the means for and demonstrate active prediction, avoidance, detection, and response to off-normal and fault events (e.g., plasma disruptions, subsystem failures): *Can the occurrence of off-normal events be reduced to levels required for power plant operation, and can reliable response algorithms be developed for acceptable device protection?*

Off-normal events will occur in any complex system. Means must be in place to detect them early and respond by minimizing and mitigating their effects, followed by recovery of control or safe shutdown, and cleanup as needed. In addition to plasma events, resulting in a rapid loss of plasma current or "disruption," off-normal events can result from loss of control, failures of sensors or actuators, or other system hardware failures. In a power producing reactor, a full unmitigated off-normal event will cause a major dump of energy on the surrounding structures with unacceptable damage. Handling of these events must be an integral part of the complete control system; it requires sophisticated control level automated decision software for determining the best course of action, as well as additional facilities for mitigation, recovery, and cleanup. Thrust 2 will develop disruption prevention and mitigation techniques for ITER, which will provide a strong foundation for this work. However, considerable challenges will remain for actively controlled, higher pressure reactor-level plasmas, and new solutions may be needed.

Specific Challenges:

In a DEMO, prediction, avoidance, and mitigation methods for plasma-induced disruptions must all be extended to greater reliability, while at the same time seeking to operate beyond passively stable limits. This Thrust will apply techniques from Thrust 2 to determine the level of performance that can be achieved through active control. For plasma-induced disruptions, the major avoidance and control tools available are the same heating and current drive actuators and feedback stabilization tools used to maintain steady state and control other transients. Much more work is needed, however, on controlling other off-normal events such as control system, sensor, actuator, or other hardware failures. New options may also exist for eliminating the plasma-induced disruptions, but there is little understanding of how these work. For example, non-axisymmetric shaping, studied in Thrust 17, and flowing liquid metal walls (Thrust 11) may eliminate or increase tolerance to disruptions, and, if results are promising, could be incorporated to enable increased performance.

Research Plan:

<u>Short-term:</u> *Implement those early detection, control recovery, and mitigation techniques for off-normal events from Thrust 2 in existing experiments that can be reliably scaled to burning plasma conditions. Initiate investigation of innovative disruption avoidance techniques.*

Medium-term: Test and optimize fully integrated comprehensive early detection and recovery control options of plasma-induced disruptions and external hardware system failures in long pulse D-D plasmas at progressively higher pressures. Consider implementation of novel disruption avoidance techniques.

Long-term: Extend the integrated control scheme for the full range of early detection, recovery without shutdown, mitigation, recovery with controlled shutdown, cleanup, repair, and reconditioning options in ITER to dominantly alpha-heated, self-sustained D-T plasmas.

Key Most Difficult Challenges

Following is a brief summary of the challenges in this Thrust that are considered most serious:

Active steady-state control: Development of diagnostics required for accurate equilibrium reconstruction capable of surviving in a high-fluence environment. Development of steady-state, high-efficiency, high-power heating and current drive systems for plasma sustainment, current profile control and MHD stabilization that can be scaled to a high-fluence reactor environment. Development of efficient actuators for controlling the density and temperature profiles in a largely self-determined alpha-heated system. Development of robust control algorithms in regulating and sustaining steady-state fusion plasmas with highly coupled, high-dimensionality nonlinear dynamics.

Startup and shutdown: Development of scenarios for startup to fully noninductive state in an alpha-dominated heating environment. These are not presently known and require study in ITER.

Burn control and thermal stability of the operating point: Development of advanced steady-state fueling techniques, for deep fueling in reactor-grade plasmas. Development of a suitable technique for selectively removing alpha particle ash.

Robust active stabilization of instabilities and transient fluctuations: Development of shielding strategies in conjunction with options for optimizing far-placed coils, or other control techniques. Internal coils and sensors for RWM and ELM control may be precluded in the high-fluence nuclear environment of a power-producing reactor, so other options need to be explored and developed.

Regulation of the power flow distribution to material surfaces: Development of a working target divertor solution that scales to a reactor is a prerequisite for developing a control system. Sufficient understanding of heat and particle transport across the plasma separatrix and into the divertor, and its dependence on the edge profiles and plasma fluctuation amplitudes and frequencies, is currently lacking.

Active prediction, avoidance, detection, and response to disruptions and fault events: Development of schemes for controlled recovery and mitigation of off-normal events that can be reliably scaled and avoid additional negative consequences. This advance would critically improve the reliability of a power producing reactor, but will require a concerted long-term research program with testing at each level.

Relation to Other Thrusts

The proposed Thrust combines many diverse elements, and key research requirements of several of the Theme 1 and 2 panels, into a single comprehensive and coordinated program where the research needs for actuators, diagnostics, and control algorithms are defined by control needs. Many of the specific elements themselves require a concerted and focused development effort, and some of these are important and large enough that they should be provided by other thrusts. Specific examples include the development of target scenarios for ITER and beyond (Thrusts 4 and 8), understanding and mitigation of plasma event induced transient events appropriate to ITER (Thrust 2), and the role of non-axisymmetric fields in improving performance (Thrust 17). Diagnostics are an integral part of plasma control. Diagnostic developments in a burning high-fluence fusion environment are the focus of Thrust 1 and strong coordination with this Thrust is required to specify those needs that are crucial for control, and incorporate realistic inputs into real-time systems. While some development of physics and actuators for ITER will be done in Thrust 4, developments for DEMO are primarily within this Thrust. Thrust 6 is expected to provide the simulation and knowledge basis for developing control-level models. Integrated demonstration of plasma boundary solutions compatible with core scenarios is undertaken in Thrust 12. Thrust 5 will in turn provide control and sustainment tools that enable the long pulse integration facility of Thrust 12. In each case, it is expected that those elements necessary for the success of the control mission will be coordinated so that solutions to specific issues found in other thrusts can be incorporated into a control system in a timely manner.

Benefit to DOE

This Thrust, if fully implemented, would have an enormous impact on the fusion program. A definitive answer to the performance limit for robust advanced tokamak scenarios will inform and have an impact on the vision of a fusion reactor, and allow comparison of the different limits and issues of other magnetic configurations. Some of the issues are generic to any fusion reactor — specifically, thermal stability and handling of off-normal events. This Thrust also has potential for a considerable number of spin-offs outside fusion: algorithms for coupling actuators and sensors in a highly nonlinear, near-marginal system with multiple time and length scales. Also, the development of new actuators — such as new radiofrequency or beam sources — needed for controlling the self-heated, self-organized plasma may have applications to other fields such as high-energy physics.

Thrust 6: Develop predictive models for fusion plasmas, supported by theory and challenged with experimental measurement

Developing the tested computational models needed for fusion plasmas will require a coordinated effort substantially beyond the current level of activities and an unprecedented degree of cooperation among program elements. This Thrust would build on a remarkable period of progress, scientific achievement and discovery, and plays to US strengths, which include the most advanced first-principles codes and the best diagnosed experiments. It leverages developments in numerical techniques and software engineering to exploit the newest generations of powerful parallel processing computers. By demonstrating deeper physical understanding of relevant science through confrontation of theory and experiments, and by developing models that embody, test and codify collected scientific knowledge, the Thrust would directly address the mission of the Fusion Energy Science (FES) program. Progress on this Thrust is of great practical importance and urgency. It would enable maximum exploitation of experiments, especially ITER, and allow for more reliable design of new experiments or facilities, critical for progress toward a DEMO.

Key Issues:

- *How well can the complex, multi-scale phenomena of fusion plasmas be understood through first-principles models, compared in detail to experimental measurements?*
- *What are the appropriate methods for integrating multi-physics and multi-scale effects, which are needed to increase the fidelity of practical computer models?*
- *How can reliable reduced, integrated models be constructed that support rapid exploration of operating scenarios and plasma control on experiments, especially ITER?*
- *What innovations in measurement techniques or experiments should be pursued that would facilitate comprehensive tests of these models?*

Proposed Actions:

- Strengthen the basic theory program to address areas where current physical models are inadequate or incomplete.
- Develop a spectrum of powerful, robust, well-verified computer models shared by a large user community. The Fusion Simulation Program (FSP), if funded beyond the program definition phase, would be an important, but not exclusive part of this effort.
- Innovate in diagnostic techniques to enable measurements critical for validation.
- Provide a spectrum of experiments including both large and small facilities, a range of confinement concepts and adequate run time dedicated to model testing.
- Conduct a rigorous set of validation activities that would assess critical elements of physical models and test them through careful comparison with experiments. These would help to guide research in theory and computation by identifying important gaps in current models.

- Recruit, train and support dedicated analysts, who would bridge the gap between theorists, code developers and experimentalists, providing unbiased assessments.
- Provide substantial computer time for code verification and model validation.

Scientific and Technical Research

The Thrust represents a focused and systematic effort to develop robust, validated predictive modeling capabilities for toroidal magnetically confined plasmas by integrating and amplifying many existing and proposed elements of the fusion program, including the FSP. The physical phenomena requiring study are broad, covering wave propagation and damping, wave particle interactions, turbulence and transport, hydrodynamic stability, plasma-wall interactions, radiation transport and atomic physics. The physical problem is intrinsically nonlinear, involving ensembles of particles and fields in a six-dimensional phase space and spanning more than a factor of 10^{12} in time and 10^{6} in space. A range of magnetic configurations must be addressed, including those with both external and self-generated three-dimensional fields.

Meeting this challenge will require significant advancements in theory and computation, along with a coordinated experimental program involving innovative diagnostics and systematic validation studies. Success would require additional resources supporting all of the actions listed earlier and more significantly, an unprecedented degree of cooperation among major program elements. Activities would be iterative and ongoing with a cycle of model development, testing and analysis at their core.

Elements of Thrust

Work on the Thrust would begin by selecting a set of “case studies” for initial attack. The case studies would represent important areas of focus that allow for rational prioritization and maximum impact on the overall fusion sciences program. Criteria for selection would include importance, urgency and readiness. One such set was discussed by the predictive modeling panel (described in Chapter 2) and is supplemented here by ideas from Themes 1, 3, and 5. These case studies, with references to technical detail in other theme chapters and sections, are provided in Table 1. They are provided only as examples; considerable discussion would be required to arrive at a final list. Available resources would dictate the feasible scale of effort in the short term — though ultimately all topics will need to be addressed.

For each case study chosen, careful planning will be carried out to map research needs and directions. Planning would begin with a discussion of how model predictions would be used, what applications are intended and what the impact of predictions would be, especially the consequences of prediction errors. Researchers would need to describe areas where the underlying physical models are well understood and where they are uncertain or controversial, to assess which parts of the problem are theoretically and computationally tractable and which require significant development. Important areas of physics integration and interface should also be identified. The current state of comparison between codes and experiments should be evaluated, and important areas of agreement and disagreement tabulated. Investigators should estimate the domain over which the model can be tested, along with the experimental platforms required. This evaluation should map out the possible domain of validation with respect to the intended applications —

will the application be an interpolation or is extrapolation required? — and consider the implications of using the model.

CASE STUDIES — SCIENCE TOPIC	REFERENCE TO TECHNICAL DETAIL	THRUSTS SUPPORTED
Plasma-wall Interactions and Scrape-off Layer	Theme 1 Theme 2 Theme 3 Theme 5	4, 5, 9, 12, 16, 17
Pedestal and Edge Localized Modes	Theme 1 Theme 2 Theme 5	2, 4, 5, 8, 9, 12, 16
Disruptions (esp. impacts, avoidance and mitigation)	Theme 1 Theme 2 Theme 5	2, 5, 8, 12
Core Transport and Magnetohydrodynamics	Theme 1 Theme 2 Theme 5	4, 5, 8, 12, 16, 17, 18
Radiofrequency Current Drive and Heating	Theme 2 Theme 3 Theme 5	4, 5, 8, 12, 16, 17, 18
Energetic Particle Physics	Theme 1 Theme 2 Theme 5	3, 4, 5, 8
Physics of Non-axisymmetric Configurations	Theme 2 Theme 5	2, 17
Integrated Modeling	Theme 1 Theme 2 Theme 5	4, 5, 8, 12, 16, 17

Table 1. A possible set of case studies for validated modeling with references to technical backup in this Report and a compilation of thrusts mutually supported by Thrust 6.

With this information in hand, physical and computational models would be developed or refined. We do not yet have all of the basic understanding sufficient to complete this Thrust, so the program will need to first identify and address missing or inadequate physics through new theoretical work focused on problems relevant to the case studies. Basic theory will also be important for interpretation of code results. Brute computational force by itself will not be sufficient, and advances in computer size and speed will continue to be needed. Innovation in algorithms and numerics for efficient solution of the computation models are critical for success, requiring expanded collaborations between fusion scientists and applied mathematicians, similar to those forged in the Scientific Discovery through Advanced Computing (SciDAC) program. A mixture of large-scale computational projects with smaller scale, more agile and speculative activities will also be essential. A particular challenge will be the integration across spatial and temporal scales and across different regions of the plasma. While work on these problems has begun, much more needs to be done to analyze the basic physics issues in play and to define appropriate methodologies for model and software module integration. This problem will be a key element in the planned Fusion Simulation Program. Fusion energy research has a long history of using computation to advance the science, so the overall challenge is to "scale up" from current efforts while building on past models of success. Required elements will include the necessary level of software engineering to make codes robust and usable beyond the core development group, and collaboration with computer scientists to define practical software frameworks, workflow schemes and data management systems.

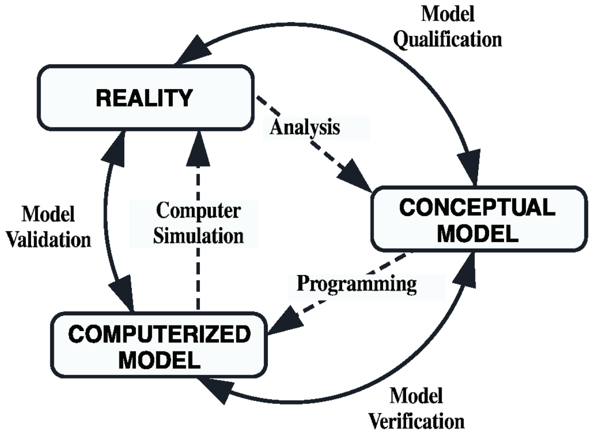

Figure 1. The processes involved in computer modeling are summarized in this diagram by Schlesinger, Simulation 32, (1979) p. 103.

A much more systematic code verification regime is envisioned as part of this Thrust. Verification assesses the degree to which a code correctly implements the chosen physical model and is essentially a mathematical problem. Sources of error include algorithms, numerics, spatial or temporal gridding, coding errors, language or compiler bugs, convergence difficulties and so forth. Methodologies for verification have been explored and documented extensively in related fields. These fall into two basic categories — those that look for problems in the code and those that look for problems in the solution. The first would be handled by traditional software quality assurance methods. The second will require testing against known analytic solutions, convergence studies and careful code-to-code comparisons. The latter underscores the importance of maintaining multiple codes, which take different approaches to solving the same problem. It is crucial to verify a code before beginning tests against experimental results — however, it is not necessary or desirable to wait until each model is "complete" before proceeding. For fusion problems, no model will ever really be complete and testing of well-verified, though imperfect, models can help guide code development by focusing on areas of disagreement.

The next element in this Thrust is the design and execution of validation experiments to assess the degree to which a calculation describes the real world. The essential activity is close and careful comparison between model output and experimental measurements, with an emphasis on quantitative measures and attention to errors and uncertainties in both code and experiment *(see, for example, P.W. Terry et al, PoP 15, 062503, 2008)*. It is a physical problem, meant to build confidence in the models and one without a clearly defined endpoint. That is, validation is not a one-time test where a code is "approved" for all time if it passes or discarded if it fails, but is instead part of an iterative process for improving the fidelity of the models. Identifying the cause of discrepancies will be among the most difficult of the challenges posed by this Thrust, but will also be among the most scientifically rewarding. For some fusion science problems, which require extrapolation past currently accessible regimes, we will need to infer the correctness of the underlying physical model to a higher degree than if we were only interpolating between accessible data points. The relation between the various processes can be illustrated in Figure 1. The first step in the validation effort, for each case study, will be the identification of critical physics issues requiring testing with due consideration to the uniqueness and sensitivity of particular measurements and model predictions. Defining measurement needs is critical and will lead to requirements for innovative diagnostics to be developed and deployed as part of this Thrust. It is highly desirable to compare predictions at various levels of integration, sometimes called a "primacy hierarch." For example, to validate a calculation of turbulent transport, one will want to compare fluctuation levels and spectra, correlations, fluxes and profiles. Thus, the diagnostic challenge will be significant and require substantial resources and attention.

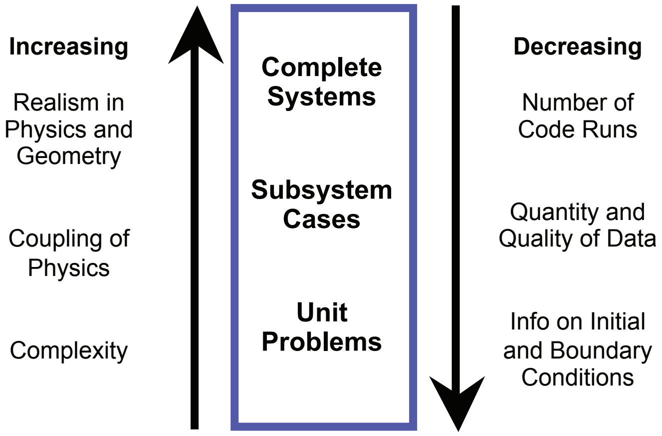

Figure 2. Validation hierarchy. Model testing is most effectively carried out by employing a set of experiments which probe the model at various levels of physical and geometric complexity.

To carry out the validation program, we will enlist a hierarchy of new and existing experiments, including a spectrum of laboratory-scale devices along with the major confinement experiments with their complement of sophisticated diagnostics. Logically, the comparisons should proceed in phases, proceeding from the simplest systems to the most complex (see Figure 2). Each level represents a different degree of physics coupling and geometric complexity. The most basic tests attempt to isolate particular physical phenomena in the simplest geometry (a "Unit Problem"). Dedicated experiments, with specialized diagnostics, will be required at this stage and some comparisons with analytic solutions may be possible. As experience is gained, researchers can be more confident when modeling more "realistic" cases ("Complete Systems"). Note that as the hierarchy is traversed, the number of code runs and experiments tends to decrease. The quality and quantity (especially spatial coverage) tend to decrease as well, while experimental errors and uncertainties increase. Information on boundary and initial conditions decreases as well. These trends suggest that at least as much attention should be paid to the lower levels of the hierarchy as to the top. These arguments may lead to requirements for new small-scale devices and/or substantially improved diagnostics on existing ones. An effort to identify opportunities at this scale should be undertaken to identify focused physics issues that could be addressed, stressing the "universality" of the issues and their relevance in fusion plasmas. A further challenge will be to provide codes that have appropriate geometry and work in appropriate regimes for each class of problems, a considerable task. On the larger experiments, which probe regimes of direct relevance to fusion energy, adequate run time and support must be made available.

The actual process of validation will be a collaboration among theorists, computationalists, and experimentalists led by dedicated analysts who are not tied to particular code development or experimental teams. These bridge the gap between specialized groups and are well placed to provide unbiased, dispassionate assessments. This role is well recognized in fields where codes of high consequence are employed. The analysts would have the primary role in defining validation tests and diagnostic needs, in coordination with modelers and the experimental teams, and would carry out much of the analysis and documentation of those tests. They would help marshal the substantial computer time and experimental run time required by the validation program. A particularly important activity would be the development of visualization tools, post processors and synthetic diagnostics to make the comparison between codes and experiments more direct and quantitative. The results of the validation experiments would feed back into the model development activities, guiding efforts to explain and resolve the discrepancies that are uncovered.

Scale of Effort and Readiness for Thrust

Significant resources would be required to pursue this Thrust with real vigor. Each element described would benefit from increased attention, better coordination and more funding. Theory is at the foundation, providing the conceptual models and mathematical formulations, interpreting the results of computations by identifying important physical processes. There will be a need for both small and large code teams, each working on problems appropriate for their level of organization. Small teams are essential for exploratory work where flexibility and agility are critical for success and where parallel efforts must be maintained. Access to specialists in applied math is beneficial to groups of all sizes, providing support for numerical methods and algorithms that can qualitatively improve computational speed. As models mature and prove their utility, greater effort is justified and a greater range of skills is required to make codes more robust and suitable for use by researchers beyond the developers. Teams, providing “production” quality tools, need resources for end-user support and training.

The fusion community needs more analysts, as defined earlier, with broad knowledge of theory, codes, experiments and statistical techniques for estimating uncertainties and errors. Computer time required for verification and validation will rival that used by the developers. And while there is a need for both capability and capacity computing, this Thrust will not be accomplished through a few heroic calculations, but by thousands of large runs, investigating parameter dependence and testing equations one term at a time.

Improved measurements are the key to validating the models and will require substantial investment in diagnostics. We will need to measure quantities that are currently inaccessible and improve the spatial coverage for those already measured. New experiments will likely be required — particularly small-scale laboratory devices that can test the most basic elements of complex nonlinear models. Advanced diagnostics will be required on these experiments, perhaps supplied through multi-institutional collaborations as they often are on larger facilities. The largest facilities will also require improved diagnostic sets and must have adequate run time to support the added demands of a more comprehensive validation program. International collaborations on experiments or diagnostics should be actively sought where appropriate.

Work on this Thrust could begin now. All of the elements described can be addressed, at some level, without waiting for future results or facilities. The FSP, which could form part of this Thrust, is already in a program definition phase. Other theory and computation activities could be augmented to fill in gaps as they are identified. Requirements for new diagnostics or new experiments could be assessed and design work begun. At the same time, this Thrust will benefit from future developments. New machines, such as the Asian superconductive tokamaks and those proposed in other ReNeW thrusts, and in particular ITER, provide platforms for model testing that extend beyond the parameter ranges accessible in current machines. Alternate concept devices would likewise extend the range for validation through tests on machines with different magnetic geometries or parameter ordering.

Integration of Thrust Elements

The elements of the Thrust combine into a unified effort, integrating resources across the fusion sciences program. Theory, computation and experiments are coordinated to solve a set of the most critical plasma physics problems. Topical science areas would be integrated at appropriate levels, spanning a wide range in temporal and spatial scales and physical phenomena. A broad spectrum of codes would be developed or exploited, ranging from single-developer research codes to large collaborative efforts, such as those produced by the FSP. The Thrust would benefit from close collaboration among applied mathematicians, computer scientists and plasma physicists. It would utilize the full spectrum of experimental facilities and require the application of new ideas and advanced technologies to produce innovative diagnostics crucial for code validation. The impact of the Thrust would be felt across the entire program, motivating and guiding theory and computation through closer interaction with experiments. It would provide stronger motivation for diagnostic development and place demands for run time on experimental facilities. Progress would enable more complex experiments to be performed with the help of accurate scenario modeling and new devices could be designed with more confidence. Improved, thoroughly validated models might allow more rapid development toward practical fusion energy, enabling larger, bolder steps. The need for improved predictive capability, to reduce the need to simultaneously achieve all reactor-like conditions, is particularly notable in Thrust 8.

The methods for coordination and management for the activities covered by this Thrust will need to be defined. Envisioned is a program where the interactions among theory, modeling and experimental efforts are stronger, more systematic and more immediate. It would probably be best accomplished through a combination of "pull," that is, a modification of the systems of rewards — funding criteria, reviews, publications and so forth — and "push," some level of centralized management and coordination. Analysts, who are crucial to this enterprise, must be trained, and stably and sufficiently supported within our existing institutional frameworks, perhaps by tying them more closely to the experimental groups. For all elements, theory, computation, diagnostics, experiments and analysis, the interdependence must be reinforced by consistent management decisions that allow for sustained progress.

Relation to Other Thrusts and Other Scientific Benefits

Validated predictive models would be essential for most or all of the other thrusts described in this Report (see Table 1). Many of these thrusts have a primary or secondary goal of improving modeling within their own domain, which could be met by a sufficiently broad definition of this predictive modeling effort. Understanding developed in these more focused efforts — notably Thrust 9 — will be integrated into more comprehensive models. The thrusts associated with ITER, and those focused on integration and control, will depend heavily on development of reliable models as will the plasma-materials interaction thrust and all of the alternate concepts thrusts. Many of the thrusts propose experimental platforms — test stands, small experiments or large devices — which would be valuable for validation of conceptual and computational models. New or upgraded facilities should be provided with sufficient diagnostics for these purposes.

Progress would be of great importance and interest outside the fusion program as well. The problems we address, like turbulence and other nonlinear interactions, remain grand challenges for physics with wide reaching implications and importance. Historically, the fusion program has been a leader in the use of high-performance computers and its researchers are already among the most successful (efficient) users of advanced computer architectures. Much of the basic plasma physics that we study has analogues in space or astrophysical plasmas or in earthbound plasmas used for industrial processes. Already, codes developed by the fusion community are in use for astrophysical problems and these kinds of interactions can only grow if we increase the scope and fidelity of our models.

Thrust 7: Exploit high-temperature superconductors and other magnet innovations to advance fusion research

An integrated program of advanced magnet R&D focuses on developing high-temperature superconductor (HTS) materials and magnet systems, which offer enormous potential for Magnetic Fusion Energy research experiments, and potentially transformational technological innovation.

Magnets are an essential, enabling technology. They confine hot plasmas and have a significant impact on the plasma initiation, heating, control, and sustainment systems. Magnetic field strength limits the achievable plasma pressure needed for fusion — higher field would allow more compact devices and could significantly ease control requirements. Today's experiments and those planned, including ITER, use superconducting magnet technology that is decades old. The superconducting magnet system of large-scale fusion devices is about one-third of the core machine cost. Future reactors must be built with the best available superconductor technology. Almost any magnetic configuration of a practical fusion reactor requires superconducting magnets.

Revolutionary new HTS materials such as Yttrium-Barium-Copper-Oxide (YBCO) are sufficiently advanced for next-step fusion applications. Besides having a high critical temperature, these materials can operate at extremely high magnetic field, offering a substantial increase in plasma performance.

Opportunity:
Success in this program can potentially revolutionize the design of magnetic fusion devices for very high performance in compact devices with simpler maintenance methods and enhanced reliability.

Key Issues:

- Development of practical conductors and cables suitable for demanding fusion applications. *Can the present fragile HTS tape geometry be integrated into high current cables with the high current density needed for fusion experiments? Can HTS material be made into round wires with high critical current density for easier magnet application?*

- Integration of HTS cables into practical magnet systems for fusion experiments. *Can HTS be used to make magnet systems with increased performance, reliability and maintainability? Which applications will most enhance performance, and reduce costs, of fusion research experiments, and ultimately enable more attractive reactors?*

Proposed Actions:

- Fabricate HTS wires, and integrate wires and tapes into high current density cables. A coordinated program of laboratory R&D in universities, national laboratories and industry.

- Develop magnet components, including improved structural and insulating elements, and assess performance for various fusion applications. Potential applications, which would greatly benefit several other ReNeW thrusts, include:

- High-field SC magnets for steady-state axisymmetric facilities with demountable joints, giving flexibility to test multiple divertor and nuclear science components.
- HTS tapes integrated into coils with complex shapes for 3-D and other alternate configurations.
- Test the most promising applications in prototypes, and ultimately incorporate into new Office of Fusion Energy Sciences (OFES) research facilities.

Introduction

New HTS materials such as YBCO offer a revolutionary path forward in the design of magnetic fusion devices that could lead to very high performance in compact devices, with simpler maintenance methods and enhanced reliability. These materials are already sufficiently advanced to consider for next-step fusion applications. The game-changing opportunities offered by these types of superconductors include the ability to optimize the magnetic fusion device for very high field plasma performance and/or to operate the device at relatively high cryogenic temperatures. They can be used with any magnetic field configuration including 3-D shaped devices. Since these materials can operate at cryogenic temperatures approaching that of liquid nitrogen (77K), one can consider as realistic the option to build electrical joints into the winding cross-section that can be connected, unconnected and reconnected in the field. The significance of this capability is that a fusion device can be more easily disassembled and reassembled to allow for easy maintenance and change of components inside the plasma vacuum vessel.

As the name “Magnetic Fusion Energy” implies, magnets are an essential, enabling technology. They are needed for confinement of hot plasmas and critical aspects of their initiation, heating, control, and sustainment. Magnetic field strength limits the achievable plasma pressure needed for fusion — higher B would allow more compact devices, or significantly ease control requirements. Superconducting magnets are required for almost any magnetic configuration of a practical fusion reactor, and the SC magnet system of large-scale fusion devices is about one-third of the core machine cost. Today’s experiments, including ITER, utilize superconducting magnet technology that is decades old. Accurate fabrication of complex magnets is also a crucial cost and performance issue for stellarators, as discussed in Chapter 5.

This Thrust responds to the new opportunities enabled by HTS materials and describes a targeted program of advanced magnet R&D that has enormous potential to enhance performance of future MFE research experiments. It addresses the key challenges of developing these materials into practical conductors and cables suitable for demanding fusion application, and integrating them into reliable magnet systems for fusion experiments.

Scientific and Technical Approach

This research Thrust will include a coordinated distribution of efforts ranging from lab-scale R&D, prototype component development, prototype magnet tests, and eventually integration into any next-step device. The actions described here will significantly expand the fusion magnet development program, which is presently modest, and engage fusion magnet experts at US universities, national laboratories, and industries. This research will require funds for procurement of HTS materials, insulation, and structural materials as well as for fabrication of components, prototypes, and the test program.

While investments made by DOE for electric utility power and high energy physics applications are yielding great progress, there are fundamental differences between these applications and fusion magnets. This Thrust would leverage this ongoing R&D, but would focus HTS magnet research specifically on the more difficult fusion magnet requirements.

High-temperature superconductors and advanced manufacturing techniques should be developed over the moderate-term with the ultimate goal of high-field operation. In the near-term, however, the properties and production lengths are now in a range sufficient for possible use in low-field fusion devices, e.g., an ST, or even with non-planar coils, e.g., helical or stellarator configurations.

Thrust Elements

A structured research and development program consists of the following elements:

1. HTS wire and tape development program.
2. High current conductors and cables development program.
3. Development of advanced magnet structural materials and structural configurations.
4. Development of cryogenic cooling methods for HTS magnets.
5. Development of magnet protection devices and methods specific to HTS magnets.
6. Development of advanced radiation-tolerant insulating materials.
7. Integration of conductor with combined structure, insulation, and cooling.
8. Development of joints for demountable coils.
9. Coil fabrication technology incorporating the unique features of elements 1-8.

Element 1 — HTS wire and tape development program

The most important goal of a high-temperature superconductor materials research program is the development of high current superconductors that can tolerate the fusion environment. This includes the production of high engineering current density tapes in long lengths. Another important path to explore is whether YBCO conductors could be manufactured as round, multifilamentary wires, more amenable to conventional methods of fabrication. Although such a breakthrough seems faraway at present, the resulting benefits would be so valuable that modest resources should be applied to address this issue. In addition, production of isotropic tapes with regard to orientation of the magnetic field is desirable.

One interesting possibility is to deposit the superconductor directly on the structural material. Presently, the Rare Earth-Barium-Copper-Oxide (ReBCO) materials are deposited by epitaxial means on the Ni-substrates with substantial load-carrying capability. If successful, this technique would obviate the need for cabling and winding.

Element 2 — High current conductors and cables development program

The goal of a high-temperature superconductor research cable program is the production of high engineering current density conductors in long lengths through cabling, bundling, or stacking of ever-larger numbers of strands until the 30-70 kA level is attained. One approach would be to develop conductor concepts such as Cable-In-Conduit-Conductor (CICC) with an adequate combination of current density, field, cooling method, and cost, but operate at higher operating temperature than present conductors.

Element 3 — Development of advanced magnet structural materials and structural configurations

Structural materials and structural concepts optimized for use with HTS material need to be developed. It is possible that conventional cryogenic materials can be used, as the heat treatment of the superconductor and the structure is not required. For cost reduction and manufacturing ease, the exploration of structural material improvements and advanced manufacturing techniques will yield quantitative reductions in magnet fabrication complexity and assembly. This is an area that has received little attention and where even limited resources may yield substantial gains.

Rapid prototyping, or "additive manufacturing," can be used to create unique shapes directly from the Computer-aided Design (CAD) models. One potential use is to manufacture the structural plates of the magnet with the features needed for operation. Multiple material deposition heads create the coil structure in a timely manner to near-net shape such as internal coil grooves and attachment features. The fabrication cost of fusion magnet structures with this technology has been estimated to be a small fraction of traditional fabrication methods.

Flexible HTS tapes integrated into conductors and grooves in structures with complex shapes could also ease the manufacture of steady-state magnets with 3-D geometry or for other alternate configurations. The 2008 Toroidal Alternates Panel recognized this as one of the most urgent issues for these configurations.

Element 4 — Development of cryogenic cooling methods for HTS magnets

Cooling methods for HTS conductors need to be investigated. Present performance of HTS materials at 77 K results in critical fields that are too low for fusion applications. The critical field, however, increases very rapidly with diminishing temperature. Alternative coolants and cooling methods, such as sub-cooled nitrogen and nitrogen-eutectics, need to be investigated. A different approach to be investigated is that of using helium gas coolant, at around 50-60 K, with conduction cooling.

Operation at higher temperatures also allows for cost savings in the cryostat, as higher heat loads can be accepted with a reduced (one-tenth) refrigeration penalty. In addition, it is possible to absorb substantially higher nuclear heating from gammas or neutrons. The heating constraints on the magnets by these processes can then be virtually eliminated. The problem of radiation damage to the superconductor and the insulation, however, still remains.

Element 5 — Development of magnet protection devices and methods specific to HTS magnets

Operation at relatively high cryogenic temperatures, e.g., in the range 40-70 K, requires reconsideration of stability, quench and magnet protection since the heat capacity of the conductors, structure, and cryogenic fluid are orders of magnitude higher than those in a magnet operating in liquid helium.

Passive and active quenching methods need to be investigated. One such method is the possibility of quenching substantial sections of the magnets simultaneously through the use of eddy current heating (or hysteresis heating of the superconductor) using radiofrequency fields. These means are not needed at liquid helium temperature because of the fast propagation of quenches, even in the presence of helium coolant. Fast quench propagation does not occur with HTS materials.

The overall design philosophy of off-normal conditions and faults also would have to be developed rigorously to guarantee protection against credible operational events. Design and analysis codes should be revised specifically for fusion magnets operating at these higher temperatures, and confirmed by comprehensive laboratory testing as has been done in the past for liquid helium cooled (LTS) magnets.

Element 6 — Development of advanced radiation-tolerant insulating materials

There has been substantial effort in the fusion community for the development of radiation-resistant insulators. Progress has been made in the development of both organic and hybrid insulators. The main characteristic of these insulators is the presence of a liquid phase that can penetrate through the coil winding, filling the voids, and impregnating the coil elements and the insulation sheets. The use of HTS can substantially change the direction of this work, opening new avenues for development of superior insulation systems. For the case of HTS material directly deposited on the substrate, it would be possible to deposit thick layers of ceramics that can serve as insulation. Ceramic insulators should survive approximately 100 times higher radiation doses than organic insulators. Means of transferring loads between plates of the magnet need to be investigated, to take full advantage of this structural potential, since the plates cannot be impregnated. The use of large plates eases the application of the ceramic insulation, with insulated windings on the plates and planar insulation between plates.

This research effort will also investigate the application of ceramic insulators to resistive coils, which may be required to be located within the plasma vacuum vessel for fast stability control of the plasma.

Although radiation damage to the magnet insulation presently limits the operating service life of the magnet system, improvements in organic and inorganic (including ceramic) insulating systems could extend the damage limit beyond that of the superconducting material, whether low-temperature or high-temperature superconducting material. At this time there does not seem to be any physical path to extend the radiation damage limit for the superconductor.

Element 7 — Integration of conductor with combined structure, insulation, and cooling

The options described above need to be integrated into a fabrication technique that takes into consideration the requirements of the superconductor, coolant, structure, insulation and fabrication. There are synergisms among these requirements that can substantially benefit fusion plasmas. The possibility of additive manufacturing, with HTS deposited on the structure, using ceramic insulation and built-in coolant passages, can substantially decrease the magnet cost while also allowing operation at higher performance (field, fusion power, pulse length). If this technique cannot be used, then methods of winding HTS in grooves on plates and insulating them need to be developed. The coolant geometry may be different in that the conductors may be able to carry the coolant themselves, as is the case with CICC.

Element 8 — Development of joints for demountable coils

The ability to operate at relatively high cryogenic temperatures and the use of relatively simple structural configurations provide very high stability, rigid operation which, in turn, allows consideration of demountable joints. Demountable high-temperature superconducting coils promise unique advantages for tokamaks and alternate configurations. They would enable fusion facilities in which internal components can be removed and replaced easily and remotely, a major advantage for the difficult challenges of Reliability, Availability, Maintainability and Inspectability (RAMI) discussed in Chapter 4.

There has been very limited investigation of demountable superconducting magnets. The use of HTS allows for relatively high-resistance joints, with modest cryogenic power consumption. The use of tapes also facilitates certain types of joints such as lap joints, where surfaces of the tapes are pressed together for a non-permanent joint. For the case of tokamaks, two types of joints can be considered, sliding joints and finger joints (fixed). In either case, it is necessary to unload the joint region, as the joint has limited load-carrying capabilities. One additional issue that needs to be addressed is cooling of the joint region. The joint region has the largest cryogenic load of the magnet, larger than leads or radiation, and it is deposited in a small volume. The joint needs to be effectively cooled. Although it is preferable to cool the joint directly, other cooling options as described should be studied.

Element 9 — Coil fabrication technology

Attractive solutions from Elements 1-8 need to be integrated and demonstrated by building prototype magnets of different configurations, e.g., planar coils, solenoids, 3-D coil geometries, etc. These must then be operated under full-scale operating conditions to the extent that they can be simulated in a prototype coil test facility. The most promising and useful magnet designs would then be incorporated into new OFES research facilities.

Scale of Effort and Readiness for Thrust

Initiation of a program of this scope will require investment of resources in funding, personnel, materials, and equipment significantly beyond those allocated to the present modest magnets base program. The HTS materials are relatively expensive at this time, and sufficient quantities of industrial quality conductor must be purchased for the lab-scale program, component development, and eventually prototype magnet development. Research and development on making advanced HTS conductors in alternative geometries requires a robust materials development program, especially for development of round YBCO wires or direct deposition of HTS materials on structures. This is also true to achieve the goals of developing structural materials with the proper alloy chemistry and manufacturing methods.

The US fusion program is ready to begin this HTS program. The remaining magnet program participants have deep experience in development of advanced low-temperature superconducting (LTS) materials, advanced structural alloys, and radiation-tolerant insulation systems. The base program experience can be applied also to component development, e.g., high current conductors, CICC, high current joints, magnet winding, insulation and impregnation, cryogenic cooling systems, and magnet structure.

Laboratory facilities are adequate to begin this program, including, for example, facilities at the MIT Plasma Science and Fusion Center and the National High Magnetic Field Laboratory. In addition, we expect to collaborate with the High Energy Physics program (e.g., Lawrence Berkeley National Laboratory) and the Applied Superconductivity Group (electric grid-based HTS systems) at Oak Ridge National Laboratory. These laboratories already have complementary HTS programs supported by DOE funding, and it would be advantageous to OFES to collaborate where feasible, leveraging these efforts and facilities.

Relation to Other Thrusts

This enabling Thrust would broaden the range of options for experimental fusion research in all Themes. For example, demountable HTS joints and/or incorporation of HTS into non-planar coils suitable for 3-D magnetic configurations would be of benefit to:

1. Steady-state integration experiments for Themes 2 and/or 3 (Thrusts 8, 12).
2. Complex magnetic configurations for Theme 5 (Thrust 17).
3. A Fusion Nuclear Science Facility for component tests for Theme 4 (Thrust 13).
4. Demountable coils that may strongly enhance the achievement of RAMI for DEMO (Theme 4, Thrusts 13, 15).

The research activities should therefore be reassessed later in the OFES strategic planning process once physics research needs have been clarified and prioritized.

Benefits for Magnetic Fusion Energy and Other Scientific Applications

Implementation of this Thrust will result in fusion devices that have high performance, high reliability, availability and maintainability with acceptable cost — potentially a “game-changer” in several respects. In the shorter term, the results of this work offer flexible experimental scale devices that can be operated in the steady state, including tokamaks, stellarators, and other non-planar geometries for 3-D magnetic configuration devices. These will enhance and accelerate the scientific research needed for MFES.

Many other scientific fields are beginning to develop HTS conductors for a variety of magnet applications, including very high field insert coils (> 25 tesla) for nuclear magnetic resonance magnets, magnetic resonance imaging magnets for diagnostic imaging and functional magnetic resonance imaging, high field, cryogen-free laboratory magnets, and possibly electric utility applications. These other applications may benefit from HTS magnet technology advances achieved under the fusion magnet program. Conversely, technology developments made for these other areas may be adapted to improve fusion magnet performance. Thus, government and industrial resources applied to HTS technology may have synergistic effects, while spreading costs and risks among different agencies and projects.

Thrust 8: Understand the highly integrated dynamics of dominantly self-heated and self-sustained burning plasmas

This Thrust explores the challenging DEMO plasma regime in which plasma self-heating (P_{alpha}) dominates over external heating sources (P_{input}), the plasma's self-generated current dominates over external current drive, the plasma pressure is high, and the plasma radiates a significant fraction of its power. In this high fusion-gain regime $Q \geq 20$ ($Q = P_{total}/P_{input}$, where P_{total} includes the energy carried by fusion neutrons) , i.e., $P_{alpha} \geq 4\ P_{input}$, the temperature and pressure profiles are set by the plasma self-heating and underlying transport processes. The transport of energy, particles, momentum, and current, and magnetohydrodynamic (MHD) processes become strongly coupled in this regime, and the plasma reaches a self-organized state, raising several new questions.

Key Issues:

- *Under these strongly coupled conditions, what is the plasma configuration that emerges from these self-consistent internal physics processes?*

- *In the strongly coupled burning plasma what maximum stability properties will the plasma access?*

- *Can such strongly coupled burning plasmas be established and sustained with much less external power and current drive than in present experiments? What is the most attractive core burning plasma regime that can be achieved?*

- *What is the self-consistent plasma core/scrape-off layer/divertor plasma state?*

ITER will provide critical information on obtaining and understanding burning plasmas. ITER's targets are to first demonstrate $Q = 10$ ($P_{alpha} \sim 2P_{input}$), in plasmas sustained by an inductive transformer (non-steady-state), and later to explore noninductively sustained plasmas with $Q = 5$ ($P_{alpha} \sim P_{input}$) for longer durations. DEMO requirements for a high-performance plasma surpass ITER's goals, and the US should explore a complementary D-T facility with a more focused physics program. The scrape-off layer plasma and its interaction with material surfaces can influence the core through particle transport, the effects of which can be observed on the multiple current profile redistribution time scale of these experiments.

Proposed Actions:

- Pursue research on existing tokamaks and the Asian long-pulse tokamaks to establish fully noninductive and high-performance plasma targets for high-Q, steady-state plasmas.

- US researchers, together with international colleagues, should assess potential operating plasma scenarios and upgrades on ITER, which could enhance the performance of noninductively sustained burning plasma demonstrations.

- In parallel, examine design options for construction of a US facility, to supplement the ITER mission, focused on high P_{alpha}/P_{input}, high pressure, high density, high self-driven current fraction, D-T burning plasmas for durations of several current profile redistribution times. This design study should explore whether the flexible facility needed for this Thrust can be made compatible with the missions of other fusion energy science thrusts.

- Based on these assessments, proceed with either ITER enhancements or a US D-T facility, or both.

The integrated core dynamics thrust is focused on the exploration of high-performance fusion core plasmas relevant to a magnetic fusion DEMO, in which the alpha power dominates the input power, the plasma current is dominated by the self-driven bootstrap current, and the strongly coupled nonlinear physics of the plasma provides a stable combination of plasma transport and MHD that allows the configuration to be sustained in steady state. The scientific challenges lie in establishing these configurations and utilizing our physics understanding to sustain and optimize them. The active control of such a plasma through fueling and pumping, heating, current drive, and rotation will be challenging and will require coupled controllers on a level not needed on present tokamaks or even ITER, to be developed in Thrust 5.

The exploration of the high-performance fusion core plasma regime is a fundamental step to verify the existence of viable plasmas for fusion power production.

1. Thrust activities

The basic parameters of interest for a high-performance fusion core plasma, based on power plant studies, and motivated by the physics requirements in Chapter 2, are shown in Table 1, along with supporting elements. These parameters must be obtained simultaneously. Key activities for this Thrust are described below, and should occur in parallel. The activities identified can benefit from activities in other thrusts, as described in the Summary.

$\beta_N^{\text{no wall}} < \beta_N < \beta_N^{\text{with wall}}$	High normalized plasma pressure
$\beta_N^{\text{ped}} \approx 0.5\text{-}1.0$	High normalized plasma pressure near the plasma edge
$3 < q_{95} < 5$	Degree of magnetic field twist near the plasma edge
$I_{\text{non-inductive}}/I_{\text{plasma}} = 1$	High fraction of plasma current not provided by an external solenoid
$0.65 < (I_{\text{bootstrap}}/I_{\text{plasma}}) < 0.90$	High fraction of plasma self-generated current
$n/n_{\text{Gr}} \approx 1$	Ratio of plasma particle density relative to an empirical limit approaching one
$P_{\text{alpha}}/P_{\text{input}} \approx 4\text{-}9$	High ratio of plasma self-generated power to externally injected power
$P_{\text{rad,core}}/(P_{\text{alpha}}+P_{\text{input}}) \approx 0.35\text{-}0.5$	Significant ratio of power radiated from the plasma core relative to the total power heating the plasma
$Z_{\text{eff}} < 2.5$	Weighted sum of ion charge in the plasma, sufficient plasma purity
$\tau_{\text{pulse}} >> \tau_J$	Plasma operation time long compared to the current profile redistribution time
Supporting elements.	
Efficient fueling and pumping, with particle control of the D-T fuel, He ash, and impurities.	
Efficient coupling of heating and current drive power into the plasma.	
Consistent pedestal density and temperature to provide high core performance with fueling and divertor compatibility.	
Multi-level feedback control on parameters ranging from plasma shape to current profile to MHD modes.	
Plasma sustainment over many current profile redistribution times (considered the longest time constants for the core plasma) without disruptions and with acceptable transients.	

Table 1. High-performance fusion core targets, and supporting elements.

In a tokamak D-T burning plasma experiment the actual sustainment of a high P_{alpha}/P_{input} plasma near the Greenwald density limit with high self-driven current would be possible for the first time. The resulting nonlinear interaction of the alpha power source with the transport, the current profile and MHD stability could be investigated, a highly critical demonstration for the viability of fusion power production. Having the sustained burning plasma, which relies on the hydrogenic ions for its power source, provides many constraints to issues ranging from fueling to alpha ash to impurity generation, which would not be present without a fusion burn. The strongly coupled MHD behavior associated with fast alpha particles, the global stability, the pedestal, and error fields, and feedback control requirements would be accessed in a D-T regime.

The longest core plasma time constant is the plasma current profile redistribution time (τ_J). The pulse lengths proposed for the Asian tokamaks, and for ITER in its steady-state mode, should exceed their plasma current profile redistribution times by a factor of at least five. However, plasma-material interaction processes can require longer times to come into equilibrium, and they may generate effects on the core over these longer time scales. The pulse length, therefore, can provide some separation of missions. To demonstrate a high-performance fusion core plasma, several issues related to the long-term plasma-material interactions may be mitigated by choosing the pulse length sufficiently long to address the core plasma processes, and short enough to reduce plasma facing component (PFC) and materials-related issues at a given step. Considerable information on handling of DEMO-level heat loads is possible on devices accessing several current profile redistribution times for integrated core demonstrations, since such a core plasma would likely deliver large power and particle loads to the divertor. To demonstrate a core plasma, it must be self-consistent with its edge plasma and plasma-material interfaces for the time scales of the core plasma experiments. Scrape-off layer plasma time scales are also shorter than the core plasma duration, allowing the observation of these physics processes. Therefore, consideration of issues related to edge plasma and material interfaces that affect the core plasma on these time scales would be both possible and necessary. Fully integrated effects of DEMO-level power densities with PFC material evolution would require longer time scales. This is an important consideration in the phasing of activities toward the fully integrated high-performance steady-state burning plasma regime.

Activity 1a: Examine potential ITER advanced tokamak (AT) scenarios in detail, with focus on making ITER more flexible in heating and current drive sources, extending to larger alpha power relative to input power, and significantly extending above the no-wall beta limit with MHD feedback control.

ITER targets are to first demonstrate Q = 10 ($P_{alpha} \sim 2P_{input}$), in plasmas sustained by an inductive transformer (non-steady-state), and later to explore noninductively sustained plasmas with Q = 5 ($P_{alpha} \sim P_{input}$) for longer durations. The later phase on advanced tokamaks, relevant to DEMO, would demonstrate a D-T plasma with a P_{alpha}/P_{input} of approximately 1, for a baseline pulse of 3000 s ($\approx$ 7-8τ_J). The normalized beta value is expected to be about 3.0. Here the plasma current would be steady state (100% noninductively sustained) with a self-driven current fraction (bootstrap) of about 50-65%. This beta and bootstrap current fraction are similar to the lower end of the projected power plant range, although at much lower P_{alpha}/P_{input}. A comparison of the ITER and power plant target parameters is shown in Table 2. Presently the heating and current drive systems include neutral beams, ion cyclotron range of frequencies and electron cyclotron range of frequencies. The use of lower-hybrid current drive (LHCD) is only considered an option. The flexibility of this mixture of sources on ITER is in question, and it is not clear that a wide range of burning plasmas can be explored, or whether steady-state scenarios can be sustained. ITER will provide an initial look at the control of plasmas with alpha power and ≥ 50% bootstrap current, along with self-consistent pedestal and edge/divertor plasmas.

	ITER-AT	**ARIES-I**	**ARIES-AT**
β_N (%)	**3.0**	**3.2**	**5.4**
$I_{bootstrap}/I_{plasma}$	**0.48-0.68**	**0.68**	**0.89**
n/n_{Gr}	1.0	1.04	0.95
q_{cyl}, q_{95}	3.8-4.5	4.4	3.0
Z_{eff}	1.4-2.0	1.73	1.83
P_{rad}^{core}/P_{input}	0.2-0.3	0.48	0.36
P_{rad}^{div}/P_{input}			0.43
$(P_{alpha} + P_{input} - P_{rad}^{core})/A_p$, MW/m^2	**0.14**	**0.45**	**0.56**
$(P_{alpha} + P_{input} - P_{rad}^{core})/P_{input}$	**1.36**	**2.52**	**6.25**
P_{alpha}/P_{input}	**1**	**3.8**	**8.8**
$\langle N_W \rangle$, MW/m^2	**0.6**	**2.5**	**3.3**
t_J, s	200-400	300	275
Duration, s	3000	~3x10^7	~3x10^7

Table 2. Comparative parameters of ITER AT and some projected power plant target plasmas.

To increase the contributions of ITER to this Thrust, the US should encourage the expansion of the heating and current drive mix on ITER, potentially including lower hybrid and upgrades of one or more of the planned sources to enable greater flexibility in producing a range of plasma configurations. The capability to significantly exceed the no-wall beta limit through MHD feedback control, including plasma rotation, should also be pursued. In addition, it may be possible to attain higher P_{alpha}/P_{input} by reducing the input power, if the plasma-generated bootstrap current can compensate for the lost externally driven current. The acceleration of advanced tokamak experiments and high P_{alpha}/P_{input} operating modes in the ITER research plan would allow more rapid identification of underlying physics for operation at high core performance. These possibilities should be assessed in detail, through predictive simulations enabled by Thrust 6, and plasma operating scenarios identified for enhanced performance in ITER.

Activity 1b: Examine design options for construction of a flexible D-T facility in the US, to supplement the ITER mission, focused on high P_{alpha}/P_{input}, high pressure, high density, high self-driven current fraction, D-T plasmas for durations of several current profile redistribution times.

The DEMO requirements for a high-performance plasma surpass ITER's goals, and the US should explore a complementary D-T facility with a more focused physics program. The US could propose to construct a more flexible D-T tokamak, optimized for the demonstration of high-performance core plasmas. The highest beta and bootstrap current fraction achieved in D-D experiments would be sought, while raising the P_{alpha}/P_{input} ratio significantly above 1, the target value for ITER, toward the Thrust mission target of 4-9. The importance of this mission is to show there is a sustainable configuration under the influence of the highly nonlinear core plasma processes. The duration of this plasma would be > 5 τ_J. The parameters and supporting elements in Table 1 would be pursued simultaneously. The US D-T facility would explore the control of core radiated power level, particle fueling and pumping, current and safety factor profile, MHD modes, and disruption avoidance in the presence of a burning plasma in a highly self-organized state. The scrape-off layer plasma and its interaction with material surfaces can influence the core through particle transport, the effects of which can be observed on the multiple current profile redistribution time scale of these experiments. In addition, issues associated with the handling of DEMO-level heat loads can be examined on these plasma durations.

Efforts would be made to keep this device smaller than projected power plants (R = 5-7 m) by focusing on advanced operating regimes with higher beta and plasma energy confinement, higher bootstrap current fractions, and higher toroidal fields, while minimizing the current redistribution time. As discussed in Chapter 2, it is expected to have gaps remaining between the simultaneous achievement of DEMO target parameters listed in Table I, and those reachable in such an experiment, although these would be minimized to the extent possible and would be bridged in conjunction with predictive simulation development in Thrust 6. This design study should explore whether the flexible facility needed for this Thrust can be made compatible with the missions of other fusion energy science thrusts, by phased upgrades and by reusing facility infrastructure and the tokamak components to the extent possible. In particular, a Fusion Nuclear Science Facility (FNSF) would require longer pulses for high fluence testing, perhaps at lower Q. Depending on its design, an FNSF could also contribute to the high-performance steady-state theme.

Activity 1c: Based on the results of the ITER enhancement studies and US D-T facility studies, proceed with the ITER enhancements or the construction of a US D-T facility, or both.

If the predictive studies carried out in ***1a*** indicate feasibility and significant benefit for steady-state scenarios, the US could support on ITER, with international cooperation, 1) the addition of lower hybrid to the mix of heating and current drive sources, 2) upgrading of existing heating and current drive sources to higher powers, and 3) upgrading the resistive wall mode coil and plasma rotation control to access higher plasma betas. Increased P_{alpha}/P_{input} above 1, in fully noninductive modes of operation, would allow exploration toward the more coupled regime of plasma transport, current and MHD under dominantly self-heated conditions. This research would be carried out in collaboration with international partners, most likely in the second phase of D-T operation.

The US D-T facility, studied in ***1b***, would perform focused research on creating high-performance fusion plasmas in the strongly coupled regime to provide the core plasma physics approaching DEMO conditions. The coupled plasma transport and MHD at high pressure, high density, and high self-driven current, exceeding that possible on ITER, would be explored. The heating and current drive sources would be used to establish these plasmas, and new and more effective control tools would be sought and developed. The DEMO parameters would be pursued simultaneously to the extent possible. Self-consistent core/scrape-off layer/divertor plasmas would be established, potentially at higher flux densities than those in ITER, and the supporting elements of fueling and pumping for particle control, power handling, multi-level feedback control with disruption avoidance would be established on the multiple current profile redistribution time scale. The studies in ***1b*** would determine the possible extensions of this facility to other fusion nuclear science missions.

2. Existing program activities that contribute to this Thrust and should be supported and expanded.

Activity 2a: Continue to pursue the program on US tokamak (and ST) facilities to establish the simultaneous high-performance plasma parameters (β_N, f_{BS}, $P_{rad,core}$, etc.) in non-burning D-D plasma.

In the very near-term, present tokamaks in the US should pursue upgrades to heating and current drive systems, as well as pulse extension to a few current profile redistribution times. These non-burning D-D plasma devices should identify attractive plasma configurations for ITER and beyond. The access to combinations of high beta, high noninductive plasma current, high bootstrap fraction, high fast particle content, long pulse lengths, other dimensionless plasma parameters, and a mix of external control tools varies among the US tokamaks. The results of the US (and international) experiments will need to be combined to establish a more universal physics basis. Using the varied tools on the US devices, the requirements for plasma current profile and radiated power fraction control can be explored (Thrust 5).

Activity 2b: Take advantage of the Asian long pulse non-burning D-D tokamaks for longer pulse lengths, all four heating and current drive sources for flexibility in plasma configurations, and control system development. Examine areas where existing device program plans for the Asian devices could be enhanced or expanded to provide a greater physics database, and seek to establish a strong collaboration to pursue this.

The main new D-D devices planned over the next 20 years are the long pulse Asian tokamaks — KSTAR (Korea), EAST (China), and JT-60SA (Japan). The focus of the Asian tokamaks is D-D operation for pulses ranging from 100 to 1000 s, which provides > 2-5 current relaxation times. In addition, the plasma configurations of interest are high noninductive current fraction plasmas (targeting 100%, and high bootstrap current fraction), and high beta pushing above the no-wall limit. KSTAR and EAST will utilize all four main tokamak heating and current drive systems (neutral beams, ion cyclotron, electron cyclotron, and lower hybrid), which will give them considerably greater flexibility than any single US tokamak in exploring plasma configurations, while JT-60SA will use neutral beams and electron cyclotron sources only. Development of more sophisticated plasma control requirements that simultaneously control several quantities can be explored (Thrust 5). Demonstration of high-performance core plasmas, for times long compared to core plasma time scales, is the major goal of these devices.

3. Summary and Linkages to Other Thrusts.

The ultimate goal of D-T plasma development is to provide the experimental validation and predictive simulation capability to project to the DEMO device with confidence. Exploring the high fusion gain core plasma regime will make a crucial and necessary contribution to this basis. The US needs to take full advantage of the burning plasma experiments in ITER, and examine how to expand these opportunities. On the other hand, the DEMO requirements for a high-performance plasma go beyond the ITER goals, and the US should explore a complementary D-T facility more focused on the physics of DEMO-relevant plasmas. Opportunities should be sought to combine or stage missions where possible to maximally utilize a US D-T facility.

Connections of this integration Thrust to other thrusts include 1, 2, 3, 5, 6, 11, 12, 13, and 17. Thrust 1 develops diagnostics for burning plasmas, while Thrusts 2 and 5 provide methods to achieve operation without disruptions or other large transient events. Thrust 3 provides, through research on ITER, insights into the alpha physics as the alpha pressure increases and the alpha heating tends to dominate the total heating to the plasma. Thrust 5 provides control approaches that allow the efficient and robust operation of the high-performance plasma sought in this Thrust. In each case, the experiments in dominantly self-sustained burning plasmas will in turn provide a key, integrated test of the capabilities and understanding developed in these thrusts. Thrust 6 provides the development and validation of predictive simulation capability, which is crucial both for projecting to the regimes of this Thrust, and ultimately to the DEMO device based on the experiments in this Thrust. Thrust 11 can provide techniques to address the power and particle handling challenges associated with this Thrust. Thrust 12 will provide insight into the integrated solutions of heat load handling, scrape-off layer plasma, and material interactions in a non-burning plasma. For expanded missions that pursue longer pulse lengths in D-T, Thrust 13 can provide solutions for a wide range of issues related to power extraction and fuel sustainability. Thrust 17 provides potential solutions where conventional tokamak approaches may be inadequate, for example 3-D fields may provide benefits in the control of transients, providing rotational transform, or avoiding the density limit.

Thrust 9: Unfold the physics of boundary layer plasmas

A thin boundary layer surrounds the hot core of all magnetically confined plasmas. The layer naturally mediates interactions between the confined plasma and material surfaces. The magnetic field structure of the region is complex. Furthermore, the plasma pressure that can be maintained at the core-boundary interface has a strong impact on fusion gain. More than a dozen important new facets of boundary plasma behavior have been discovered over the past decade. Despite this progress, the basic processes that determine the local spatial scale lengths, and the heat and particle flow within the layer, are still not adequately understood. Hence, the heat and particle loads on plasma facing components, impurity intrusion, and core fusion gain are difficult to predict, making design requirements and operational strategies uncertain and necessarily conservative. The output of Thrusts 9, 10, and 11 will be an essential design component for any new fusion device and can be extensively tested in future facilities (e.g., Thrusts 8, 12, 13).

Key Issues:

- Only a part of the physics controlling the boundary layer has yet been identified. *How can we fully identify and characterize the physics controlling the boundary layer and resulting plasma-wall interaction (PWI) sufficiently for physics-based scaling to future devices?*
- Models to predict the complex features of the boundary layer are immature. *How can we accurately describe the highly turbulent boundary layer plasma with material erosion in comprehensive simulations to create simplified models?*
- Specifications for active internal components, such as radiofrequency antennas and launchers, and passive diagnostics, are limited by our ability to predict plasma fluxes to those components, and erosion caused by the radiofrequency interaction with components at remote locations. *How can the predictive capability of plasma edge modeling, including material interaction with internal components, be improved?*
- The existing ITER design is projected to have little margin for managing the plasma heat load, and higher-power devices will require substantially increased power exhaust requirements. *How can the magnetic configuration of the boundary region be modified to spread out the heat flux at the material interface?*

Proposed Actions:

- Develop and deploy new diagnostics in existing devices for comprehensive boundary layer measurements of plasma flow, density, temperature, electric field, turbulence characteristics, and neutral density in at least two dimensions and, as appropriate, three dimensions, to provide the data necessary to uncover the controlling physics.
- Increase the level of effort on validation of individual edge turbulence and transport codes, then expand this effort to involve more comprehensive boundary layer models.
- Develop measurements and predictive capability of the plasma fluxes to radiofrequency antennas and launchers; develop models for the self-consistent modification of the boundary layer plasma by the radiofrequency wave injection and other internal components.

- Design and implement innovations of the boundary magnetic geometry in existing devices to demonstrate optimized plasma heat exhaust that is within material limits, and design and implement such a configuration in a future fusion device.

Summary

This Thrust focuses on the physics of the boundary layer, an essential area for fusion development where understanding is incomplete. Advancement for this area requires development and implementation of both a substantial set of new edge diagnostics and increasingly realistic models, which together can lead to identification of the dominant governing physics and to predictive capability. As peak heat fluxes to materials projected for DEMO are unacceptable, we propose implementation and testing of new boundary magnetic field shapes that may reduce peak heat fluxes and also improve edge-plasma characteristics.

Introduction

A thin boundary layer surrounds the hot core of all magnetically confined plasmas. The layer encompasses the interface between adjacent regions where magnetic field (B) lines are closed (the "pedestal") or open (the scrape-off layer [SOL]), and it mediates interactions between the hot confined plasma and material surfaces that receive the plasma exhaust and generate wall impurities. Furthermore, the plasma pressure that can be maintained at the interface with the core plasma (the pedestal) has a strong impact on fusion gain. Despite the layer's importance, the basic processes that determine its spatial gradient lengths, and the heat and particle transport within the layer, are complex and not adequately understood. While substantial progress has been made, with more than a dozen important new facets of boundary plasma behavior having been discovered over the past decade, our understanding of the physics is significantly incomplete.

The complexity of the boundary layer arises from features that make one-dimensional (1-D) diagnosis inadequate and direct application of many core-plasma models inappropriate:

- Interacting gas (plasma + neutrals), photons, solids, and sometimes liquids.
- Steep gradients, increased collisionality, and transition from closed to open magnetic field lines yield strong 2-D (sometimes 3-D) profile variations. In contrast, core profiles typically only vary radially (1-D) owing to rapid transport along the B-field.
- The boundary magnetic field exhibits strong shearing as the field lines change from a closed to open topology that is characterized by very long, thin and twisted geometry.
- Plasma fluctuations, often intermittent, exceed 10% of the background, and can approach unity, which invalidates the small-amplitude diffusive transport model.
- Large electric fields near radiofrequency antennas and launchers, and on surfaces connected to antennas along the B-field, can result in greatly amplified plasma sheath potentials and enhanced ion sputtering.

Boundary Characteristics and Issues

Plasma boundary layer physics involves the interaction of a number of key phenomena that can bridge adjoining subregions. The boundary layer begins in the closed field line region where the maximum plasma pressure obtainable, and thus often core performance, is set by a combination of plasma radial scale lengths and magnetic field shear structure. Excessive pressure results in a well-documented magnetohydrodynamic (MHD) instability — the Edge Localized Mode (ELM) — that causes periodic, rapid local ejection of plasma into the SOL. The ELMs thus limit core performance and present potentially damaging impulsive heat loads to material divertors and walls. Edge localized modes are a major concern for ITER and DEMO, and must be strongly mitigated. The physical understanding used to predict ELM amplitude and frequency, and to devise mitigation strategies (see Thrust 2), rely on identifying the plasma and neutral transport and turbulence processes that determine the pedestal structure (local gradients); despite progress, no first-principles model exists. Also, shaping of the magnetic equilibrium to access more benign ELM regimes should be tested.

The boundary layer must also distribute the steady-state heat exhaust to wall components. The power handling capability of divertor and first-wall surfaces cannot be directly scaled up to match the corresponding scaleup in burning plasma power output. Steady-state power of ~10 MW/m^2 remains the practical divertor limit with 1-5 MW/m^2 on some walls; ITER is being designed for such limits. Steady-state heat removal poses operational constraints (e.g., requiring detached divertor plasma operation). Lacking careful control of steady-state operation and ELMs, divertor and first-wall components will be destroyed, and ITER's scientific mission would not be realized. In light of this expectation, it is not yet possible to design a credible DEMO-class fusion device, with a size like ITER but power output that is ~4-5 times higher, whose material surfaces can survive the expected heat exhaust. Specifically, predictive understanding is lacking for boundary layer heat and particle transport dynamics, which typically involve filamentary structures; yet this physics defines not only the level of plasma-wall interaction (peak heat flux, particle fluxes, erosion rates), but also the boundary conditions imposed on the core plasma (edge gradients, flows, impurity levels).

The transport simulations to model plasma fluxes use ad hoc radial transport coefficients, rendering them more in the class of "interpretive," rather than predictive simulations. This limitation prevents fundamental prediction for heat-flux widths and pedestal/SOL transport at present, although it does test parallel (along **B**) transport models. Impurity transport in the pedestal/SOL is treated at the fluid level through the multi-species 2-D transport codes or by trace-impurity Monte Carlo ion codes, using fluid or experimental background plasma for the hydrogenic species. Limited code validation with data often shows rough agreement with various quantities for "attached" divertor plasma conditions, but typically only to the 30-50% level. For "detached" conditions (planned for ITER), the disagreement is worse. There appears to be a larger-than-predicted plasma density in the private-flux region that may have an important effect on divertor behavior. These problems likely need refined models of neutral transport physics and radiation trapping.

For turbulence models, there is a set of 3-D fluid codes that likely contain important electromagnetic effects, one of which spans the separatrix. These models describe some of the dominant characteristics of the measured fluctuation features (some spectra similarities, blob-like propagation, intermittency), but detailed validation is sketchy (large fluctuations and transport increasing with density). However, at a fundamental level, there is not yet a clear consensus concerning the controlling instabilities and saturation mechanisms. In contrast, core turbulence simulations, theory and experiment have standard benchmarks (e.g., the so-called Cyclone test case for turbulence-driven transport), agree on many aspects of the big picture of the dominant ion transport, and have made good progress on electron transport.

In addition to radiative dissipation and detached divertor operation, further reduction of the peak heat flux to divertor surfaces may be obtainable by modification of the details of the boundary magnetic configuration. One recent idea of this approach is the Super-X divertor where the SOL magnetic flux tube is extended to substantially larger major radius to increase the area available for heat deposition and increase the effective distance between the core edge and the divertor plate. A second idea, the Snowflake divertor, expands the flux tube locally to spread the heat flux, expand the effective core/divertor distance, and increases magnetic shear that can affect ELM stability. Initial modeling results are encouraging for both of these ideas, but detailed experimental tests are required. There is clearly a range of configurations intermediate to the Super-X/Snowflake that should be examined, including careful evaluation of the minimum practical angle between the divertor surface and B that would also improve conventional divertors. Another method for increasing divertor heat-load capabilities involves various PFC materials and is discussed in Thrust 10.

As the plasma has an impact on the material surface, it causes heating, absorption and recycling of hydrogen, and sputtering of surface material; in some circumstances, melting of the surface can occur. The recycled and sputtering neutrals enter the plasma, where they are ionized and become a plasma ion species responding to the electromagnetic fields. Many of the ions are returned to the surface in a process called redeposition and can thus build up layers of new surface material that may be an elemental mixture of different portions if the walls are made of several materials (e.g., carbon, beryllium, and tungsten mixing in ITER is a poorly understood issue). Establishing the basic science of these processes as verified in linear plasma simulator devices and beam-particle test stands is being proposed through Thrust 10. First-principle molecular dynamics codes can treat mixed materials, but the key input of the inter-atomic potential is largely unknown even for carefully prepared samples, let alone the mix that develops in an operating tokamak (see Thrust 14). In this Thrust, the near-surface plasma modeling needs to be combined with comprehensive SOL modeling in a time-dependent manner to properly model the erosion and buildup of new material layers.

A special plasma-wall interaction that requires careful evaluation and integration is the interaction of the edge plasma with radiofrequency antennas and launchers — devices used extensively to inject auxiliary power into fusion devices to heat and control the core plasma. Large electric fields and sheaths can be driven at the antenna and launcher or where the B-field line touching the antenna also comes in contact with a remote material boundary. The sheath potential can accelerate ions into the materials, yielding undesirable sputtering of metallic impurities and power loss at the antenna. Other parasitic radiofrequency losses in this region include edge modes (shear Alfvén and cavity modes), parametric instabilities, and nonlinear wave-particle interactions. The radiofrequency sheath potential can also drive radial E×B convection in front of the antenna, which increases the radial flux of plasma to the wall. Models of the radiofrequency sheath require treatment of both the ion and electron Debye length space scales, either explicitly, or by a sheath boundary condition, which is nonlinear.

As a final note, the overall importance of the boundary layer is underscored by the European tokamak program's fundamental reorientation toward boundary research over the past decade. An outcome of this shift is the decision to dedicate Europe's two most important tokamaks to exploring materials other than carbon; JET is committed to studying a beryllium-wall plus tungsten-divertor PFC system, while AUG is focusing on an all-tungsten PFC system. Thus they have decided that enhanced tokamak core physics priorities be subordinated to learning how to tame the plasma-materials interface. The Europeans have also been steadily growing their boundary modeling and diagnostic efforts. The present US situation stands in significant contrast: staffing levels in US tokamak scrape-off layer diagnosis and physics research have decreased over the past decade from about 40 to about 20 people working at the major devices. The US fusion program cannot afford to fall behind in this critical area, recognized by the Greenwald Report as being at the top of the Magnetic Fusion Energy priority list. At a minimum the heavy investment in the existing major US devices needs to be better exploited for addressing critical boundary issues. This will require restoring, at least, the earlier level of effort as well as major investment in edge diagnostics.

Proposed Actions:

1. Develop and deploy new diagnostics

Understanding complex systems requires extensive diagnostics. For example, exploiting aerodynamic lift and drag for aircraft design would have been impossible if the air velocity were measured at just one or two locations around airfoils; such measurements are typically made at dozens of locations. The pedestal/SOL is much more complicated than air flow around an object and one can hardly expect to identify edge controlling physics with the spatially sparse diagnostic deployments available today. In addition to the diagnostics sets described below, other techniques should be assessed.

Extensive sets of pop-up (or similar drive) probes should be used to map out the 4-D (spatially and temporally) distributions of density, electron temperature, and electric potential (Langmuir probes); parallel flow velocity (Mach probes); cross-field transport (turbulence probes); and ion temperature and energy distributions of electrons and ions (gridded energy analyzers). Magnetically activated probe drives of various types have been developed, including swing action and reciprocation, achieving viable penetration to the separatrix. Deployments of such probes are needed in significant numbers (e.g., > 10 systems), around the periphery of tokamaks. While materials costs will be relatively modest, the additional staffing requirements will be significant.

Thomson scattering is complicated in edge plasmas by high background light, but it has been successfully used on DIII-D to reconstruct the 2-D profiles of electron density and temperature (n_e and T_e) by sweeping of the magnetic X-point relative to the Thomson channels. With the addition of a high-dispersion instrument such as a Fabry-Perot interferometer, measurements could be made of electron/ion energy distributions and Z_{eff}. In addition, a multi-laser Thomson system (firing laser 1, 2, 3 in rapid succession) could make images that reveal the detailed time evolution of so-called "blobs" and ELMs. Charge-exchange recombination (CER) spectroscopy using new powerful pulsed ion diode neutral beams (10^6 A/m^2, 1 μs, 100 keV/amu) may permit measurement of T_i and $v_{||}$ (along $\mathbf{B}$) in to the separatrix. Advances in ultrasoft X-ray spectroscopy would allow reconstruction of the T_e profiles with time resolution higher than possible from Thomson scattering alone.

Cross-field transport in the pedestal/SOL has to be better characterized to understand the physics controlling pedestal structure and its stability, and also to be able to scale PWI at the main walls or at the divertor plates for plasma being lost from the pedestal/SOL region. With regard to wall erosion, the problem can be unwanted material being transported to the divertor rather than excessive erosion of the divertor material. Wall particle and power fluxes need to be measured systematically in three dimensions (toroidally, poloidally, temporally) as functions of plasma parameters to identify the scaling parameters. The underlying pedestal/SOL turbulent plasma transport mechanisms responsible for these spatial distributions need to be established through turbulence measurements, including 2-D imaging, and power spectra (both frequency and wavelength-resolved) of density, temperature, potential and magnetic field — ideally measured at the same time so that their phase relationships can be understood.

Real time, in situ surface diagnosis of H-uptake, erosion, deposition, and co-deposition is required. Such surface studies are seriously compromised today because most of the experimental information is global: post mortem analysis provides spatial resolution but integrates over entire campaigns, while other studies are for single shots but integrate over all internal surfaces. It is difficult or impossible to extract the controlling physics from such integral experiments. Real time, or at least between shot, in situ surface diagnosis is required, e.g., using an on-site accelerator whose probing ion beam can be brought to various locations inside the vessel using the tokamak coils.

2. *Validate existing codes and extend models*

Basic understanding and predictive capability for guiding device operation and designing improved plasma facing components and new machines require detailed models that include the relevant physics. Modeling the boundary plasma can be divided into describing how (a) the plasma power and particles coming from the core region are distributed to PFCs, (b) the material is modified by such bombardment (e.g., sputtered or recycled particles, tritium transport and retention), and (c) the erosion products in turn modify the SOL and core plasmas. While important progress has been made, each area needs a substantial new effort in model validation and model development. As more experimental data becomes available, new and likely unexpected features will be revealed that will further motivate upgraded models. Simulation tools in these areas are all ready to advance, but are budget-constrained.

a. Pedestal and SOL models: Simulation models of the edge and SOL include transport codes, which give the slow evolution of the plasma profiles and fluxes in a complex environment, and turbulence codes, which model the unstable, strongly fluctuating plasma state to (ideally) provide transport coefficients to the transport codes. This latter connection has only been made for a limited set of simulations. The edge transport codes are either 2-D fluid or now emerging 4-D (2 spatial coordinates, 2 velocity coordinates) kinetic. The turbulence codes are 2-D and 3-D fluid and emerging 5-D gyrokinetic codes.

A focused effort is required to better model the often dominant plasma turbulence in the pedestal/SOL region, using existing fluid and developing kinetic codes. The associated understanding and quantification of underlying edge transport mechanisms need to be brought to at least the level of that for core transport. Given the edge's importance and complexity, this will require a substantial investment. This work should include the nonlinear saturation of ELMs and how they eject plasma from the pedestal, through the SOL, and onto PFCs. There should be a major effort to verify (code-theory and code-code) and validate existing fluid turbulence codes with the existing and extended diagnostics. As the kinetic codes become more capable, they should be used to quantify kinetic corrections. Important basic theory work that should be done includes unstable modes in the complex edge geometry, development of validated reduced models, and a set of gyrokinetic equations that are implementable in numerical codes. Consideration should also be given to full three-velocity simulations that don't require a gyrokinetic approximation.

Boundary transport models that describe both pedestal structure and heat-loads to PFCs need to be enhanced to realistically include time-dependent, large-amplitude filamentary structures from short wavelength turbulence-driven "blobs" to large ELMs. Options to be examined further are efficient reduced-dimension models (3-D to >2-D), a time-averaged coupling to turbulence code, and direct turbulence and transport simulations for long-time transport. Impurities should be included to understand their transport through the time-dependent SOL. Kinetic edge codes have begun to quantify neoclassical ion transport in the edge more carefully, a key addition that should be supplemented by reduced models that can provide faster fluid transport models. As with turbulence models, an extensive verification and validation program should be undertaken for transport models (see also Thrust 6).

b. Near-surface models and utilizing wall response models: The plasma and neutral energy and particle fluxes incident on PFCs cause sputtering, recycling, and in some extreme cases, melting of the materials. The removed material is ionized within the plasma and often flows back to a nearby surface location in the process of redeposition, which can have major impact in fusion devices, including tritium retention. The wall response is incorporated at a basic level in the SOL transport models through recycling and sputtering boundary conditions, but these now lack kinetic detail and time-dependent coefficients.

Presently, these recycling and sputtering processes are treated in a simplified manner by whole edge transport codes, while detailed guiding-center or full 3-D orbit Monte-Carlo codes are used in the near-surface region. Although some limited file-based coupling of these two models (whole edge and near-surface) has been undertaken, a more vigorous program is warranted. The coupling should be more tightly integrated (no human intervention) to allow iteration for self-consistency and time-dependence. This would describe the nonlinear sputtering and recycling responses to blobs and ELMs, as well as long-time changes associated with wall temperature changes and particle saturation. The details of the sheath can be calculated from a 3-D PIC code, but again needs integration into other components.

There is a strong connection to Thrust 10 aimed at developing models of the surface response and validating these models in linear plasma "divertor simulator" devices and beam-particle test stands. These models are needed here to provide the basic data on particle recycling and sputtering. Furthermore, changes in the surface-material properties are closely related to the response data needed here and that aspect of the issue is included in Thrust 14.

The modeling of near-surface plasmas for liquid surfaces uses similar tools and has similar issues to the modeling of solids, but here the evaporative fluxes into the plasma volume can be important. Furthermore, the motion of the liquid is an issue that has received some attention, but needs more work for flowing-liquid systems (also see Thrust 11). Given the encouraging results on existing devices, the edge modeling capability that includes liquids, while available in some codes, should be expanded.

Dust formation and transport are a major concern in future large devices because of their potential mobilization during an accident and possible core impurity contamination. Models of dust transport exist and can be utilized. Needed are better understanding and models of dust formation, either in Thrust 10 or Thrust 14.

c. Integrated models and synergies: The couplings between the dynamic wall and the SOL, between the SOL, pedestal and core, and between local turbulence and long-time transport are not well modeled. This is a key area for continued development, with initial Scientific Discovery through Advanced Computing (SciDAC) projects now underway with FACETS for the core/pedestal-SOL/wall and the Center for Plasma Edge Simulation (CPES) for the pedestal/SOL (see also Thrusts 4, 6). The development of models associated with the boundary region should be designed in standard manner to fit into integration efforts. Important questions that can be answered by such coupling include the following:

- How does a dynamic wall/SOL/pedestal system respond during time-dependent “blob” or ELM transport, i.e., how is it different from time-averaged models?
- How do SOL/pedestal turbulent transport, neoclassical transport, and particle sources combine to determine the pedestal plasma and divertor heat-flux profiles widths?
- How do impurities transport into the core through a turbulent SOL/pedestal?
- How is the core density limit related to pedestal/SOL behavior?
- How do SOL flows influence core rotation?

3. Diagnostic and model development for radiofrequency antenna and launcher interactions
As noted, a substantial number of boundary plasma simulation codes and radiofrequency codes exist, but these two communities have not collaborated strongly. Thus, there should be a concerted effort to integrate radiofrequency effects into boundary codes and vice versa. There is also only sparse data on radiofrequency sheath interactions leading to sputtering, and this should be enhanced.

To complement the outline of the types of boundary tools available above, the radiofrequency SciDAC project, including collaborators in Europe, is developing a set of radiofrequency codes to describe the antenna coupling (TOPICA) and wave propagation (AORSA, TORIC). Some preliminary work has been done on the use of a sheath boundary condition in these codes and a newly developed code to describe radiofrequency sheaths in the SOL plasma. This work is guided by a number of models to describe radiofrequency sheath formation under various assumptions. Some studies of radiofrequency-induced sputtering and self-sputtering have also been carried out and compared with JET and TFTR data. Very modest work on physics integration has been carried out. Incorporation of a SOL wave-sheath code is necessary for quantitative work. Edge physics transport and turbulence codes are available in which radiofrequency effects can be included.

The action items are as follows: (1) Use improved diagnostic coverage to verify the strength and location of radiofrequency sheaths, confirming connection along **B** to the antenna. The associated sputtering should be quantified by spectroscopic measurements. (2) Compare and verify existing radiofrequency sheath models; aid inclusion in boundary codes. (3) Improve or develop models for parasitic power loss from radiofrequency antennas and launchers; develop methods for reducing the loss. (4) Incorporate radiofrequency models in boundary codes and validate with diagnostic data. (5) Incorporate radiofrequency sheath and loss models in antenna and propagation codes. (6) Integrate boundary transport and turbulence, wall codes, and radiofrequency antenna and propagation codes to understand the interdependence (e.g., gas puffing to enhance radiofrequency coupling to the main plasma).

4. Integrated experimental tests and development of innovative divertors

While this section is brief, it nevertheless involves a major effort in testing and utilizing the building blocks of Sections 1-3. Complete physics understanding and validation of models in all boundary areas discussed will require significant operational time on confinement experiments. While dedicated test stands can better address validation of certain elements of the models (see Thrust 10), more dedicated run time will be needed on existing machines to properly test integrated performance in a realistic environment.

Innovative divertor magnetic configurations such as the Super-X and/or Snowflake for reducing peak heat fluxes offer high payoff and should be tested. These configurations can be formed in existing devices without new magnetic coils. Some Snowflake configurations have already been made. The Super-X may require installation of some extended divertor surfaces at larger major radius to accept the heat flux. Promising results of such tests should be followed by extensive analysis to optimize their implementation in future devices, and to ensure that all the functions of a divertor — power exhaust, acceptable impurities, and helium exhaust — can be performed simultaneously in the very challenging environment of future fusion plasmas. Other innovations related to divertor materials are given in Thrust 11.

All operational scenarios and new device designs discussed in ReNeW will require substantial input from the knowledge gained in this Thrust, to ensure that the boundary layer plasma will satisfactorily perform the many functions required for high-power fusion devices. While the emphasis here has been on tokamak boundary plasma, every effort should be made to understand the range of device types where the data and models are applicable and how to extend them where needed.

Thrust 10: Decode and advance the science and technology of plasma-surface interactions

Plasma-surface interactions define the boundary condition for any magnetically confined plasma. Many improvements in core plasma performance have resulted from changes in the way the material surface responds to the incident plasma. Improvements in wall preparation techniques have optimized performance in present-day confinement devices, but the reasons for this success are poorly understood. Extrapolation of plasma-surface interactions from today's pulsed devices to future steady-state machines (i.e., with hot surfaces under intense plasma and neutron flux) provides little confidence in predictions of future core plasma performance.

Key Issues:

- Plasma-surface interactions vary by orders of magnitude depending on temperature, incident species, surface material and exposure time. *Can we reliably extrapolate conditions at the wall of today's pulsed confinement machines to future steady-state reactors?*
- Transient expulsions of energy from the edge plasma cause unacceptable erosion of existing wall materials. *Can we develop clever new concepts to extend the wall operational limits?*
- Surfaces evolve when subjected to plasma, neutron and edge alpha irradiation. *Is it possible to predict the impact of this evolution on an equilibrium plasma state and on plasma facing component lifetime during steady-state operation? Can more resistant materials and coatings suitable for use in diagnostics, or high-power radiofrequency and microwave components, be developed?*

Need for Dedicated Facilities:

Dedicated new and upgraded facilities are required to evaluate the impact of intense plasma flux, transient plasma events, long pulses, and elevated temperatures on the performance of surface materials and designs. Since many of the required assessments do not need the full complexity of a toroidal magnetic confinement device, dedicated laboratory experiments using linear plasma devices can provide a well-diagnosed, well-controlled, and cost-effective environment for performing plasma surface interaction research. The effect of extended pulse lengths and elevated wall temperature on many plasma-surface interaction processes can be addressed more quickly in these devices. The fundamental science advances from this Thrust will be key in the design of plasma facing components and internal components for long pulse, hot wall devices, e.g., Thrusts 12 and 13.

Proposed Actions:

- Upgrade existing laboratory facilities and test stands, and build new facilities capable of extending plasma-surface interaction parameters closer to conditions expected in fusion reactors, including the capability to handle tritium, liquid metals, and irradiated materials.

- Build large-size test stands where full-scale internal component tests and design validations can occur.
- Develop and improve first-principle models of the material and plasma coupling for future fusion machines by validating against new experimental data.
- Invest in surface material diagnostics to quantify material behavior and evolution.
- Develop and test new surface materials to improve performance margins.

Summary

To address knowledge gaps in the plasma-surface interaction (PSI) for DEMO, extensive experimental and theoretical studies are required. This Thrust describes an effort of enhanced modeling, combined with measurements from both existing and new dedicated PSI test stands, that employ simple geometries for high-throughput, well-diagnosed material characterization. Strengthened effort in these theoretical simulations, validated properly by data from test stands, can significantly increase understanding and predictive capabilities of all the PSI processes that affect both plasma facing components (PFCs) and internal components (IC). Advances in plasma-material interactions (PMI) research through these new and existing test stands will not only prepare for DEMO, but will provide new and exciting opportunities for interactions with the broader materials research community.

Note that this Thrust focuses on the first few micrometers of the materials, whereas Thrust 11 focuses on bulk material innovation and characterization. Also, the research from this Thrust will contribute strongly to and benefit from the enhanced understanding of boundary plasmas in present devices (Thrust 9) and in a future high-power device (Thrust 12), as well as the nuclear science of Thrust 13. In addition, the PSI characteristics of some of the advanced materials developed in Thrust 14 would be tested here.

Introduction

Plasma-surface interactions define the boundary condition for any magnetically confined plasma. As device size and power increase, the limitations to operating regimes become dominant. In ITER the operating scenarios are already constrained to limit the heat flux to plasma facing armored surfaces, internal control coils are being added to control intermittent edge localized mode (ELM) events, and concerns have been raised about the heat loads delivered to antennas and launchers. The functional lifetime of the diagnostic structures necessary to monitor and control the plasma, and the objects adjacent to them, are uncertain. In larger, higher power, steady-state DEMO-class devices, the operational window may be vanishingly small or nonexistent. These imposed limitations are partially due to an incomplete understanding of both the underlying physics and technology governing the coupling between the plasma and these adjacent objects.

All present US confinement facilities use inertially cooled objects surrounding the plasma. This relatively low-tech approach is suitable because the devices all operate in a short-pulse (< 10s) fashion. Pulsed operation also allows for conditioning of the plasma facing surfaces between discharges, and manages PSI for transient achievement of preferred confinement regimes. The technology encounters new challenges with the use of actively cooled components and longer-pulse (up to 1000s) operation in ITER. Further challenges are presented when addressing the higher power density, high-temperature wall, bulk (neutron) and surface (charged particle) accumulated damage and near steady-state operational conditions necessary for a Fusion Nuclear Science Facility (FNSF) or DEMO device.

PSI Research and the Need for Advanced Test Facilities

Ultimately, the control of PSI must be demonstrated in a fully integrated confinement facility. However, much of the understanding necessary to demonstrate control of PSI can be achieved through the use of lower cost, more flexible, and partially integrated test facilities.

Examples of the PSI knowledge gaps that can be addressed utilizing PSI test facilities are as follows:

- *Erosion and redeposition* — Knowledge gaps remain in the basic processes that govern erosion; an example is the impact of mixed materials on chemical sputtering. In addition, the DEMO PSI will involve thick layers of redeposited material, whose interaction properties are as yet unknown. The ability to predict the erosion and redeposition of material, including the synergy of multi-species plasmas and mixed materials, is required.
- *Tritium retention* — Tritium control and accounting will be crucial for the success of DEMO. To achieve this, the processes that lead to retention of tritium in materials subjected to high plasma particle flux, and the feedback of this saturated surface with the incident plasma, need to be understood.
- *Radiation transport* — The conventional divertor scenario results in an optically thick divertor at DEMO parameters. For prediction of divertor operation under these conditions, plasma and neutrals models with radiation transport and trapping need to be validated with high-quality data.
- *Transient events* — The survival of plasma facing materials to thermal transients will set the limits on acceptable size and duration of the transient events. The damage to materials caused by both rare, large events and frequent, smaller events must therefore be well characterized.
- *Testing of new materials and concepts* — The extreme environment expected in DEMO will require new material developments and concepts. Exploration of materials and the characterization of promising candidates need to be performed to test their possible application to DEMO.
- *Neutron irradiated materials* — The high neutron fluence expected in DEMO will affect erosion and tritium retention properties of materials in components. Near-surface material properties will be altered by the neutron damage. Thus, characterization of neutron-damaged specimens and validation of performance are needed.

The scientific bases for filling the knowledge gaps in PSI technology can be accomplished by using dedicated test facilities. At the simplest level, ion beam facilities allow measurements of the elementary processes of chemical and physical sputtering, in an environment that allows full control of the energy, impact angle, and species of bombarding ions onto well-controlled and characterized target surfaces. More sophisticated single-effect science ion-beam facilities equipped with in situ surface diagnosis can study more complex dynamic mechanisms such as surface mixing, morphology evolution and phase-dependent erosion, providing surface-response code validation. Facilities that perform plasma bombardment of surfaces allow the study of synergistic effects that are limited using ion beam experiments. These plasma experiments allow macroscopic PMI processes to be studied, such as erosion, redeposition, and hydrogenic retention, and can validate models of these processes, albeit not in the full DEMO plasma regime. A combination of ion-beam analysis in situ during plasma-material exposure provides direct measurement of plasma dynamics and surface evolution during exposure, and is essential to PSI basic science and development. As part of this research Thrust, resources for both existing ion beam and plasma bombardment facilities should be increased to enhance predictive capability and understanding on PSI issues.

Further synergistic effects are expected to be important in the DEMO PSI environment that cannot be addressed in the experiments described, motivating a new class of facility. For example, the erosion and redeposition process will ultimately be determined by the properties of redeposited layers. To simulate this, experiments need to be performed in the "strongly coupled" regime where eroded material is ionized in the plasma and returned to the surface. This requires a facility with characteristic dimensions larger than the neutral mean free path, which equates to a large plasma cross-section operating at very high density ($>10^{20}$ m^{-3}). Furthermore, achievement of erosion conditions relevant to DEMO requires particle fluxes greater than 10^{23} ions/m^2/s and the ability to run for extremely long pulses, which further enhance the redeposited layers. Very high density is also required for model validation in optically thick regimes to address the gap in radiation transport. Assessment of retention of tritium and helium at DEMO-relevant parameters requires the ability to test materials at very high temperatures (> 600 ^oC). This can be readily achieved in the presence of very high plasma heat flux, which would also enable this device to perform tests of materials for PFCs and internal components. An important capability would be the qualification to handle neutron-irradiated samples. The neutron damage that will be present in DEMO will have a profound impact on near-surface and bulk materials properties, affecting, for example, tritium retention as well as heat transfer and material joints in internal components. The ability to measure these effects with rapid turnaround of experiments is needed.

The new class of facility described can be characterized by its ability to fully simulate the DEMO divertor. Such a facility should have the following characteristics, which would fill many of the gaps requiring an integrated effects approach:

- Particle flux $> 10^{23}$ $m^{-2}s^{-1}$, and target heat fluxes up to 20 MW/m^2, delivered to target materials with inclined surfaces operating at elevated temperatures (> 600 ^oC).
- Large plasma area, ~100 cm^2, to produce a strongly coupled plasma simulating detached conditions in a divertor. This will allow for the testing of small components and multilayered materials used, for instance, in antenna structures and microwave launchers.
- Ability to handle hazardous materials, such as beryllium, and neutron-irradiated material samples with significant displacement per atom.

- Ability to handle liquid metals, e.g., gallium or tin, as well as more reactive liquids, e.g., lithium.
- Variable magnetic field, with maximum $|B| > 1$ tesla, to study PSI effects in the divertor region.
- Progressively longer plasma durations, starting from 10^2 s, and eventually to 10^6 s, respectively corresponding to the anticipated time constants of the physical processes of high-Z impurity migration, to hydrogenic species permeation in ferritic steel under relevant operating conditions.
- Full suite of diagnostics for PSI measurements, including in-situ optical and mass spectroscopy, laser-induced fluorescence, thermal desorption spectroscopy, quartz-crystal microbalances, as well ex-situ analysis of the target by UV and visible Raman spectroscopy, scanning and Auger electron microscopy, X-ray photoelectron spectroscopy, direct-recoil spectroscopy and atomic force microscopy. Measurements of plasma density, total ion flux, neutral density, and electron and ion energies will also be required using interferometers, Langmuir probes, retarding potential analyzers, spectroscopic diagnostics, etc.
- These experimental measurements would be invaluable for the validation of the computational simulations of the plasma-surface interactions using a multi-scale approach, which will be key for the successful prediction of DEMO PSI.

As indicated, to evaluate the performance of surface materials and designs from the impact of transients, long pulses, and elevated temperature, dedicated new and upgraded facilities will be required. Many of the required assessments do not need the full complexity of a tokamak. Dedicated laboratory experiments can provide a cost-effective, well-diagnosed, long-pulse, and well-controlled environment for performing PSI research. These facilities should be operable in pulsed mode to simulate disruptions and ELMs to understand the effects of these phenomena on target materials over longer times than are achieved in today's confinement devices. Similarly, the effect of extended pulse lengths and elevated wall temperature on many PSI processes can be addressed in simplified linear plasma devices on a shorter time scale.

Although reliable PSI performance will need to be ultimately demonstrated in a long-pulse, high-power density, high wall temperature, toroidal confinement device, the knowledge basis developed in this research program should establish confidence in the success of these integrated demonstrations. Moreover, the knowledge gained here will guide the design of such future facilities. The fundamental science advances from this Thrust will be key in the design of plasma facing components and internal components for long pulse, hot wall devices, e.g., as envisioned to fulfill the goals of Thrusts 12 and 13.

Predictive Modeling, Validation, and Experiment

A strong theoretical basis is required to further the understanding and prediction of the PSI process in magnetically confined fusion plasmas. Therefore, an integral part of this Thrust is for enhanced modeling combined with measurements from both existing and new, dedicated PSI facilities that take advantage of simple geometries to enable high-throughput, with well-diagnosed material characterization. Modeling efforts will span a range of complexities and assumptions, from exploration of elementary processes such as sputtering due to single-ion impact, to coupled codes modeling the full PSI environment. These include, for example, sheath formation, erosion and material migration, and evolving surface morphology and concentrations of mixed materials. Strengthened effort in these theoretical simulations, validated properly by measurement from test facilities, can significantly increase understanding and predictive capabilities of the PSI processes.

Internal Components and Radiofrequency Testing Needs

Functional internal components (antennas, launchers, sensors, mirrors, control coils, etc.) must meet the same criteria of other plasma facing components in terms of resistance to high heat (~1 - 20 MW/m^2) and neutron fluxes and acceptable levels of impurity production, while also maintaining the capability to perform heating, diagnostic, or control functions. Internal components can also suffer damage if they lie in the path of particle fluxes produced in transient events; an important example of such a scenario is the production of runaway electrons during a tokamak plasma disruption. There are significant challenges in the areas of diagnostics, radiofrequency antennas and launchers, and internal control coils.

Burning plasma properties introduce new fundamental measurement limitations to some existing measurement techniques, and present an environment that challenges a range of diagnostics. Maintenance of the optical quality of mirrors, polarizers, and windows located close to a burning plasma environment represents a significant challenge. In particular, redeposition and erosion combined with long plasma exposures are significant concerns.

Reliable performance of radiofrequency antennas and launchers is needed. Many of the issues concerning the interaction of the near field of the antenna and launcher with the scrape-off layer plasma (SOL) are not well understood. There is a need to predict, measure, adjust for, and modify this edge environment since it is the region through which power is coupled. Radiofrequency breakdown and arcing in the antenna structure are some of the main power limiting issues with operating the antenna in the plasma environment, and are poorly understood. In addition, the antenna structure and Faraday shield will likely be constructed from layered or coated materials that require good conductivity and high heat resistance. The behavior of these structures in a nuclear environment and the survivability in long-term operations are concerns.

Performance validation of ICs is required from both dedicated test stands and testing on toroidal facilities. While many of the PSI processes can be addressed in a linear plasma device, the performance issues for antennas and launchers can be addressed in a specialized radiofrequency test facility. Dedicated test facilities offer the advantages of providing a controlled and flexible environment that is well-diagnosed and capable of long-pulse and cost-effective operations. Toroidal devices can be used for integrated effects, model verification, and demonstration of fully reliable IC components.

A dedicated high flux linear facility can be used to validate many IC design issues, including thermal stresses at the joints subjected to high heat and neutron fluxes, cracking due to voids in brazes, tritium transport, erosion of coatings, insulator performance, and similar issues. A dedicated radiofrequency test facility can be used to develop and validate antenna performance issues and radiofrequency sheath formation. Arcing and breakdown issues can be addressed where multiple parameters can be controlled and tested for long-pulse operation. Other issues that can be addressed include simulation of the antenna interaction with the SOL, including radiofrequency sheath dynamics, antenna phasing effects, hot spot formation, localized erosion, transport along and across magnetic field lines, and wave-particle interactions. Many aspects of edge modeling can be tested and validated. A dedicated radiofrequency test facility will allow for easy access and rapid changes in antenna and launcher geometry or test conditions, and can be used to develop new diagnostics, control models, and antenna and launcher concepts that can be operationally verified before implementation on a confinement experiment.

Advanced Materials for Fusion Devices and Relation to Materials in Thrust 14

The material systems comprising the plasma facing components of present-day fusion devices, and those envisioned for ITER, share a common limitation. Given our current understanding of the extreme irradiation environment of next-generation reactors, none appear capable of withstanding the extraordinary combination of ion fluence and bulk neutron damage expected. Moreover, the plasma-surface boundary plays a key role in fusion performance of the core plasma by the self-consistent recycling of particles. Therefore, materials must address two primary design constraints: 1) maintain a viable structure in an aggressive nuclear environment and 2) maintain a viable plasma-surface interface compatible with burning plasma operation (e.g., steady-state, high temperature, high edge alphas fluence, high displacement per atom levels, high-intensity transients) that introduces minimal contamination via erosion, redeposition and recycling processes at the plasma edge. Much of the fundamental materials science in terms of surface evolution under ion irradiation, diffusion of hydrogen species under non-equilibrium irradiation conditions, and the neutron-induced evolution of microstructures in advanced alloy systems is not yet fully understood. There are fundamental physical limits challenging the current suite of materials.

As an example, if one considers the three candidate materials typically considered for structural and plasma facing application — graphite, beryllium, and tungsten — each has a finite lifetime based on either PMI compatibility or structural integrity dictated by plasma-surface coupling, neutron exposure and operating temperature. In the case of graphite, anisotropic crystal growth leads to unavoidable materials disintegration. In turn, graphite suffers from hydrogen-dominated chemical erosion with strong temperature dependence. Neutron-irradiation-induced low-temperature embrittlement or high-temperature swelling limits beryllium as does its high hydrogen retention properties. In the case of tungsten, locked defects are created directly within the neutron cascade which, at least at moderate temperatures, leads to severe embrittlement and causes fracture toughness and flaw tolerance to plummet. Furthermore, edge alpha-induced surface morphology protrusions (e.g., “fuzz”) can introduce intolerable high-Z amounts into the boundary plasma, leading to poor confinement.

To this point, research and development for both the plasma facing materials and the divertor structure have been somewhat limited to the issues relevant to current machines. To address the significant challenges of future fusion power devices, a more robust coupling of fundamentally based materials development with advanced design is required. Specific areas of study would include PMI testing of adaptive materials compatible with coupling of the plasma-surface boundary and an aggressive nuclear environment. Concepts can include low-Z/high-Z alloys or nanocomposites (e.g., boron or silicon with tungsten), solid flow-through materials (e.g., carbon) and refractory ultracrystalline thin-film coatings. Furthermore, needed are a detailed mechanistic understanding and development program for refractory materials including tungsten, extending the irradiation and temperature performance of advanced nano-composite steels, and a definition of the absolute design lifetime limits of copper alloys.

Such a materials-science-based development needs to be closely integrated with a vigorous material design effort, which is the role of Thrust 14. Careful coupling of the material design effort with material testing will be required to advance a science-based development approach. The changes and interaction of the newly proposed materials will be examined for compatibility with the strongly coupled plasma-surface boundary in this Thrust. Such testing will need to proceed in a timely and cost-effective manner to ensure that the results are incorporated into improved material designs that will enhance the performance of the developed materials. Furthermore, design advances will require materials that are radiation-tolerant and provide adaptive materials surfaces that can operate in a burning plasma environment. Providing a facility that can both test for plasma and radiation (i.e., neutrons, etc.) damage will provide a unique opportunity to test advanced materials under simulated, burning plasma reactor-relevant conditions.

Facilities for examining materials exposed to high levels of neutron exposure are limited. For the high-dose, high-temperature materials of particular interest here, the lack of a high-flux 14 MeV neutron source will prove a critical obstacle. While ion beam and fission reactor sources of damage are currently available and should be exploited, neither are able to yield the appropriate ratio of atomic displacement damage to transmutation helium production that will occur for the sample sizes required for meaningful component characterization. Therefore, options for a large-volume, fusion-relevant neutron source should be evaluated.

Relationship to Ongoing PSI Research

This R&D approach has the potential of establishing a broad understanding and predictive capability of the fundamental processes and their synergistic interactions, which in turn advances theory, modeling and simulation of these processes in fully toroidal configurations. This approach for PSI research, on the other hand, does not intend to address physical phenomena related to the interacting toroidal plasma outside the last closed flux surface in the toroidal configuration.

The new PSI facilities envisioned here will complement and extend the capabilities of current PSI research facilities, provide new research opportunities to the broader materials community, and promote US competitiveness in this area. These facilities are foreseen to be user facilities whose design will be developed through PSI and fusion community participation. The upgrade of domestic testing facilities, and the education and training of operating personnel, are critical to ensure the development of a sound PSI basis for DEMO.

Thrust 11: Improve power handling through engineering innovation

Plasma facing components (PFCs) in a power reactor will receive high heat and particle fluxes from the plasma and will require active cooling. Water cooling technology used in ITER is inapplicable to a reactor that will operate with high-temperature solid walls or reactive liquid metals. For DEMO, either solid PFCs (cooled by high-pressure helium or liquid metals) or free-surface liquid PFCs (such as lithium or tin) could be used. Both PFC innovations will be developed for acceptable power and particle handling.

Key Issues:

- Higher heat removal requirements are anticipated for DEMO due to the higher power and particle fluxes to the divertor and first wall, which must operate at elevated temperatures for optimum power conversion efficiency. *How do we develop better PFC designs that operate at higher temperatures and can remove higher heat loads with adequate design margin? Can we exploit larger area targets with smaller inclination angles (<1°) and obtain better alignment to the magnetic flux surfaces?*

- Our ability to design and fabricate PFCs for high-temperature operation is limited by the brittle nature of refractory metal alloys in the case of solid PFCs and by our lack of knowledge of magnetic forces and temperature limits on free-surface liquid metals. *How do we develop new low-activation alloys in the case of solid PFCs and assemble the necessary database for liquid PFCs? Is it possible to develop new materials that are radiation tolerant while having minimal impact on the plasma during quiescent and transient plasma operation?*

- We do not understand the mechanisms of tritium permeation through PFC materials to the coolant, including the effect of neutron damage. *How do we study the synergistic effects of irradiation damage and tritium permeation for solid and liquid PFCs?*

Proposed Actions:

- Design, fabricate and test refractory heat sinks with advanced cooling techniques for high-temperature operation (>600C) and deploy liquid metal PFC experiments in plasma devices.

- Develop fabrication processes and better joining techniques using reduced activation refractory alloys for both PFCs and internal components, e.g., radiofrequency launchers.

- Construct and upgrade new lab facilities for synergistic testing including cyclic high-heat-flux, irradiation and permeation, and liquid metal performance; and improve models of thermal performance, irradiation damage and tritium transport in PFCs.

- Provide improved PFCs for qualification on existing or new confinement experiments.

- Develop more robust PFCs for transient events with higher design margins and improved reliability and maintainability. Include engineering diagnostics to monitor PFC performance and provide data for lifetime prediction models.

Introduction

Thrust 11 addresses the fundamental engineering science and the technology required to develop components that can handle the higher heat fluxes in a power-producing fusion reactor. Much of this technology is useful in alternate concepts as well as conventional toroidal devices. Plasma facing components such as those in the first wall and divertor, and Internal Components (ICs) such as radiofrequency launchers, Faraday shields and electron cyclotron resonance heating (ECRH) mirrors, will need to operate at higher temperatures and handle particle and heat loads much higher than ITER. First wall and blanket components may consist of refractory alloys, vanadium alloys, reduced activation ferritic martinsitic steel (RAFMS), oxide dispersion strengthened ferritic steel, and silicon carbide composite. Power conversion systems for DEMO will require heat sinks for plasma facing components that can operate at high temperatures as part of a highly efficient closed Brayton cycle. Plasma facing components could endure peak heat fluxes in excess of 10 MW/m^2 at the divertor and localized peak heat fluxes as high as 0.5-2 MW/m^2 on the first wall and internal components. Since these components are an integral part of the power conversion system, this work must connect closely to Thrust 13 to ensure careful integration with blanket requirements. The technologies and components developed in this Thrust feed directly into Thrust 12, and likewise benefit from the integrated testing in Thrust 12's toroidal confinement facility. In addition, the activities in Thrust 11 focus on the bulk PFC properties and complement the basic plasma-surface interaction research of the first few micrometers described in Thrust 10.

A close coupling must exist between the PFC materials research focused on the development of more ductile refractories and improved joining techniques in Thrust 14 with PFC development in Thrust 11. Both free-surface liquid metal PFCs and ducted liquid metal and gas coolants in solid PFCs are under consideration. Currently, the Technology Readiness Level (TRL) (Mankins, 1995) for helium-cooled components is approximately 3 and that for liquid metal PFCs is about 4. More fundamental research is required to assess the temperature limits of free-surface liquid metal PFCs and devise efficient power conversion systems that address this limitation. Certainly, further development is required to reach the technology readiness level required for DEMO (9-10). Progress is stymied by a lack of investment in high-temperature materials and liquid metal research, and the limited capabilities of our present test loops and equipment to operate at the required high temperatures and pressures. Unless the community addresses this need in a timely manner, we will not have the actively cooled components required for high-temperature applications like DEMO, or even the intermediate facilities needed to fulfill the goals described in Thrusts 12 and 13.

Advanced Cooling Technologies for Solid PFCs

Helium cooling has many advantages in a nuclear system due to its inherently safe, inert chemical properties; lack of corrosion; vacuum compatibility; single-phase heat transfer without the possibility of a critical heat flux (CHF) excursion; lack of neutron activation; and easy separation from tritium. Most importantly for DEMO, helium is the fluid of choice for a highly efficient, high-temperature Brayton cycle exhibiting minimal wear and corrosion of gas turbines, and the closed cycle helps contain any tritium reaching the coolant.

One key disadvantage of helium is its low thermal mass, ρC_p, that is less than 1% of water. This necessitates the use of high mass flow rates and greatly enhanced heat transfer area and turbulence promoters for efficient heat transfer. Tremendous progress occurred in this regard during the last fifteen years. Small device testing demonstrated that helium cooling can be as effective as water when handling several tens of kilowatts. Other disadvantages for helium and the Brayton cycle include the nature of a compressible gas working fluid that requires higher pumping or blower power and larger supply and return piping compared to liquids, and the safety issues surrounding the large amount of stored energy in high-pressure systems.

US research efforts on helium-cooled heat sink applications began in 1993 on copper devices. Small businesses fabricated helium-cooled heat sinks with various enhancement techniques including microfins, jet impingement, plasma-sprayed, sintered or brazed porous media, and metal foams. Several of these devices exhibited record heat flux handling capabilities with heat transfer coefficients close to 30,000 W/m^{2}K as shown in Figure 1. Refractory heat sink development started in 1998 on pure tungsten and continues today with lanthanated tungsten, tungsten-rhenium alloys and molybdenum (Youchison 2001, Diegele 2003).

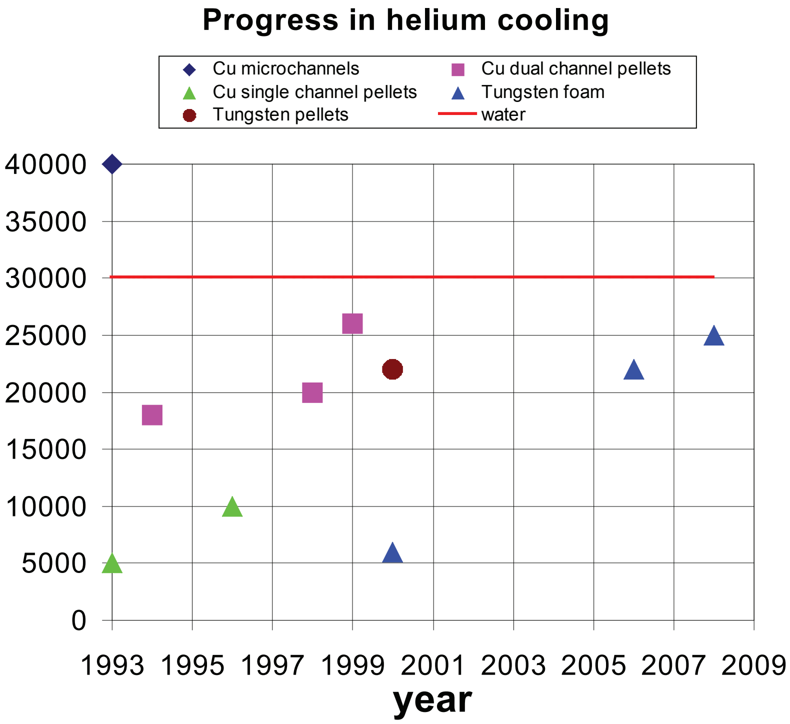

Figure 1. Helium now rivals the cooling capability of water on small devices.

Consider pure tungsten as an example. Tungsten heat sinks require inlet temperatures greater than 600 °C (DBTT) and must operate at the highest outlet temperatures possible (~1000 °C) for higher efficiency, but stay below the recrystallization temperature (~1100 °C). Gas cooling studies, including thermomechanical modeling and high heat flux testing, and substantial upgrades to test facilities, are necessary to evaluate these components. The production of reliable high-performance heat sinks for DEMO PFCs will require further refractory materials development, innovative fabrication techniques and clever thermal engineering coupled with power-relevant testing.
US industry was instrumental in developing low-cost, near-net-shape fabrication techniques for our small, present-day helium heat sink mock-ups for first wall, divertor and IC applications. The

technology is now evolving to include large area (0.3m x 0.3m), multiple-channel components with integrated manifolds and a minimum of joints. Successful demonstration and further scale-up are required. Another possibility is the creation of monolithic PFCs consisting of refractory heat sinks and refractory armor such as tungsten rods or lamellae with no joints or thermal expansion mismatches. In other cases, diffusion bonding may be appropriate, for example, where refractory armor joins to oxide dispersion strengthened ferritic steel transition pieces followed by RAFM steel heat sinks. The Japanese recently demonstrated diffusion bonding of tungsten to RAFM steel for high-temperature blanket applications.

Thrust 14 will address radiation damage and tritium permeation through PFC materials in the presence of a neutron flux and fluence. This information is necessary to affect design features that help minimize the tritium inventory in the coolant and compensate for degradation in thermal and mechanical properties in the PFC.

Advanced techniques for robotic nondestructive evaluation (NDE) require further development to ensure the fabrication of robust PFC joints including welds, brazes, and diffusion bonds that can transfer the heat in the presence of high edge alpha and neutron flux. In addition, NDE techniques are required for performance and maintenance evaluations during reactor operations. Selected PFCs must include integrated engineering diagnostics such as flow, temperature and stress measurements to monitor run-time performance. These engineering diagnostics — coupled with routine NDE during maintenance periods — will supply the data necessary for the development of PFC lifetime prediction models. Such models are needed to formulate preventive maintenance protocols.

Vertical heat removal targets with <1° inclination and careful alignment techniques will be pursued to take better advantage of new divertor configurations, like the contoured surface, Snowflake or Super-X divertor, developed in Thrust 9. Any thrust in innovative power handling with solid PFCs must include development and testing of a variety of heat sinks. This includes application of sacrificial low-Z coatings on PFC armor, and the development of ductile refractory coatings and structural materials undertaken in Thrust 14. It also includes thermomechanical modeling, high heat flux testing of large panel, helium-cooled heat sinks with integrated manifolding and diagnostics, and the inclusion of a complete helium heat-transport system. This will enable researchers to carefully evaluate the influence of mass flow rate, pumping power, operating system pressure, residence time, and flow instabilities due to nonuniform heating on the thermal performance and reliability of the components. In addition, single and multiple effect tests and integrated test results can validate the thermal modeling and identify failure mechanisms. Both porous media and new jet impingement devices, developed through international collaborations, are prime candidates for heat removal in actively cooled PFCs. Other concepts, including internal surface roughening, swirl tubes and 3-D fins, are also appropriate. Further R&D at the national labs, universities and industry is necessary to optimize and select a reliable, high-performance system.

Since the current level of development has outpaced the capabilities of DOE to test the devices under DEMO-relevant power and flow conditions, we must include new and upgraded coolant loops to complement the test stands in Thrust 10. DEMO-relevant helium flow loops should be capable of high-temperature operation (~1000 °C), 10 MPa of operating pressure and provide mass flows on the 500 to 1000 g/s level.

Liquid Metal Plasma Facing Components in Burning Plasma Reactors

As discussed in the preceding sections, refractories and other solid materials require substantial development for use as a PFC in a fusion reactor. Operational limits on solid PFCs, at present, constitute one of the most significant restrictions on design space for the intermediate device, DEMO and follow-on fusion reactors. However, in addition to solids, there are several candidates for liquid metal-based PFCs, including gallium, tin, lithium, and tin-lithium eutectics. Among these, lithium and probably the tin-lithium eutectic should provide a low recycling surface, while other liquid metals are high recycling. A flowing liquid metal PFC would have limited residence time (~tens of seconds) in a fusion reactor before removal and recirculation. Hence, erosion, helium and neutron damage, and tritium retention are not significant issues (provided that low recycling liquid metals, such as lithium, can be adequately purged of tritium before recirculation). Plasma material interaction issues (sputtering, evaporation) are now limited to the liquid metal PFC, whereas the solid substrate supporting the liquid only sees the neutron damage. The separability of PMI and neutron damage considerably simplifies material qualification for reactors. The possibility of using thin layers of liquid permits intensely cooled systems, with the plasma-exposed surface closely coupled to the underlying coolant (either helium or a flowing liquid metal).

However, liquid metal PFC development is in an early stage. There are very few, relatively small, liquid metal PFC test facilities in the US. Only a few liquid metal systems have been tested in tokamaks, with a focus on lithium as a tool to reduce hydrogen recycling and high-Z impurities. The implementation of liquid metal PFCs in tokamaks has been predominantly in static or evaporative systems (Majeski 2006). Additional tokamak devices (FTU and T-11M) have demonstrated withstanding heat loads above 5 MW/m^2 (Vertkov 2007) using capillary porous systems (CPS) for liquid lithium PFCs. Much higher power loading (>50 MW/m^2) has been demonstrated in evaporatively cooled test stands, but not in tokamaks. The use of fast flowing liquid metal jets, for example, has been tested in only one or two very small devices.

Prominent issues for both high and low recycling liquid metals include the entire problem of introducing the liquid metal to, and removing it from, the reactor, and inducing stable flow to transport the fluid from inlet to outlet. Magnetohydrodynamic (MHD) effects caused by the excitation of electrical currents in the liquid metal PFC must not cause macroscopic influx of the liquid metal into the plasma. Sputtering and evaporation must be kept to acceptable levels including temperature-enhanced erosion, and this dictates the temperature limit for the coolant. Heat removal must be effective below these temperature limits and be compatible with the power conversion system. Coverage of the underlying substrate by the liquid metal, in the case of slow flow, must be complete, since the substrate will not be designed for exposure to plasma. For jets or open-surface channel flow, splashing and surface variations must be eliminated. For capillary systems, clogging and nonuniform coverage must be avoided. The design of inlet manifolds and fluid collection systems is a challenge for either type of system. Tritium migration through the liquid metal into underlying coolant channels must be investigated; since different liquid metals have differing affinities for hydrogen, this work is specific to each candidate liquid metal and eutectic. Finally, for lithium, the physics consequences of low recycling walls for tokamak equilibria must be thoroughly explored, since the consequences for reactor design can be considerable. This last issue closely and explicitly links liquid metal PMI and the fusion core.

The elements of a thrust to develop liquid metal PFCs are also important for liquid metal flows in the blanket ducts contained in Thrust 13. They include:

1. Magnetohydrodynamic modeling of liquid metal transport at high Hartmann and Reynolds numbers with fusion relevant fields, configuration and magnetic field gradients, including the effect of plasma MHD on the stability of liquid metals. This activity is a combination of theory, simulation and small laboratory experiments for validation.

2. A multi-laboratory effort to investigate substrate optimization (for slowly flowing liquids), including capillary effects, as well as general chemical effects, temperature limits, corrosion (including corrosion of neutron-irradiated materials), wetting, etc., is needed.

3. Design and engineering of practical devices for injecting, controlling and removing liquid metals in the presence of fusion-relevant magnetic fields.

4. Design and engineering of systems to remove heat from liquid metal PFCs, for slowly flowing liquid metals. Fast flowing liquids (e.g., jets) would carry the heat load with the fluid.

5. A comprehensive liquid metal test stand capability. An adequate test stand should include a liquid metal loop feeding and draining a target surface, with appropriate flow rates over the surface, at a relevant magnetic field, and field angle to the surface. High heat flux testing using e-beams or lasers should be available. The ability to perform simultaneous PMI studies, at least on a time scale appropriate to the residence time of the fluid on the target surface, is desirable. Comprehensive diagnostics of the liquid metal surface behavior should be available.

6. Following test stand qualification, a plasma confinement device should test these liquid metal systems with discharge durations in excess of the residence time of the fluid in the system.

7. Demonstration of high-power conversion efficiency using removed heat under DEMO-like conditions.

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Thrust 12: Demonstrate an integrated solution for plasma-material interfaces compatible with an optimized core plasma

The plasma facing components (PFCs) in a DEMO reactor will face much more extreme boundary plasma conditions and operating requirements than any present or planned experiment, including ITER. Solutions to make these boundary requirements compatible with a sustained core plasma in a regime attractive for fusion need to be validated under DEMO-like conditions. The goal of this Thrust is to integrate the plasma-material interactions thrusts (9-11) and control thrusts (2, 5) and demonstrate their compatibility with a high-power efflux, steady-state, optimized driven core plasma.

Key Issues:

- Power density a factor of four or more greater than in ITER. *What techniques for limiting both steady and transient heat flux to surfaces are compatible with optimized core and boundary plasma performance?*
- Continuous operation resulting in energy and particle throughput 100-200 times larger than ITER. *How can material erosion, migration and dust production from plasma facing surfaces be made compatible with long-term operation and core plasma purity?*
- Elevated surface operating temperature for efficient electricity production. *How can plasma facing materials and configurations be optimized at elevated temperature for fuel recycling and plasma interactions?*
- Tritium fuel cycle control for safety and breeding requirements. *How can hydrogenic fuel retained in materials be reduced to acceptable levels?*
- Steady-state plasma confinement and control. *What techniques are simultaneously compatible with core plasma sustainment and edge plasma power handling requirements?*

Proposed Actions:

- Develop design options for a new facility with a DEMO-relevant boundary, to assess core-edge interaction issues and solutions. Key desired features include high-power density, sufficient pulse length and duty cycle, elevated wall temperature, as well as steady-state control of an optimized core plasma. Hydrogen and deuterium operation would allow flexibility in changing boundary components as well as access for comprehensive measurements to fully characterize the boundary plasma and plasma facing surfaces. The balance and sequencing of hydrogen and deuterium operation should be part of the design optimization. Develop an accurate cost and schedule for this facility, and construct it.
- Extend and validate transient heat flux control from Thrust 2, plasma control and sustainment from Thrust 5, boundary plasma models from Thrust 9, plasma-material interaction science from Thrust 10 and plasma facing component technology from Thrust 11 with the new research from this facility, thereby demonstrating a viable solution to the challenging core-edge integration problem for DEMO.

The plasma facing components in a DEMO reactor will face much more extreme boundary plasma conditions and operating requirements than any present or planned experiment, including ITER and the superconducting tokamaks abroad. The gap between current and planned tokamaks and DEMO in the area of compatible core-edge solutions is very large in terms of steady-state handling of DEMO-level power efflux on high-temperature material surfaces. This gap needs to be bridged in a manner consistent with external control of plasma current, heating, and fueling to maintain an optimized clean, hot and dense core plasma. The configurational flexibility and complete diagnostic coverage that will be needed for testing solutions for this demanding environment would be most readily accomplished in a limited activation facility. The knowledge gained from this research program would accelerate the development of integrated core-boundary solutions for high-fluence burning plasmas such as in a Fusion Nuclear Science Facility (FNSF), and a demonstration (DEMO) fusion power plant.

Solutions for individual challenges outlined should be addressed by several thrusts to inform the design of the facility to support Thrust 12. The physics of the boundary layer plasma and methods to disperse the heat flux over a greater surface area are to be investigated in Thrust 9. Physics of the plasma surface interaction are to be investigated in Thrust 10. Heat removal technologies and components will be developed in Thrust 11. The effects of neutrons will be tested in Thrusts 11 and 13. Techniques for limiting heat flux transients will be developed in Thrust 2, and methods for core plasma sustainment and control will developed in Thrust 5. However, there is no present or planned facility where these solutions can be combined and the complex interaction of these issues studied and tested at DEMO-like conditions: *this is the central goal of Thrust 12*.

Primary Challenges for a DEMO Boundary Configuration

The most basic of physical conditions that a DEMO fusion power plant must accommodate is the very high heat efflux from the burning plasma core that arrives on PFC surfaces. For typical DEMO designs, the total power flowing that must be exhausted onto material surfaces is approximately four times that predicted for ITER, yet in a device of similar physical dimensions. Since ITER's PFCs are already at the limits of engineering design, new techniques for spreading the exhaust heat flux over as much of the vessel surface as possible must be developed for DEMO.

Simultaneous with such high-power handling requirements, DEMO will have a much higher duty factor (≥80%) than existing or planned devices, resulting in an annual energy and particle throughput 100-200 times greater than for ITER. Such constant high loading would lead to macroscopic levels of erosion of PFC material, which will be thin (~ 5 mm) for reasons of heat removal and tritium breeding. An associated concern is that the eroded material may redeposit with poor thermal and mechanical properties.

While various aspects of DEMO heat flux solutions will be developed in Thrusts 9, 10 and 11, the PFC surface designs and operational techniques must be shown to be compatible with a robust edge pedestal that assures high core confinement and simultaneously allows efficient core control techniques. Steady-state operation will require external means of driving plasma current, fueling the core plasma and, ideally, optimizing the core plasma profiles. An important core-edge issue is the efficient coupling of the control carrier (e.g., waves) through a "thick" edge and collisionless pedestal into the core. We must also demonstrate that the required control launchers can be designed and operated to accommodate a high heat flux environment with high first-wall temperatures.

Another aspect of control developed in Thrust 2 that must be shown to be compatible with the DEMO heat flux environment is the control, avoidance, and mitigation of heat flux transients such as those from edge localized modes (ELMs) and disruptions. The energy released in an uncontrolled DEMO ELM is likely to be ~ 4 x that of ITER where control techniques are already required to reduce the energy released in an ELM by ~ 90%. Disruptions as well must be essentially eliminated in a DEMO. Techniques for avoiding, or suppressing, these transients must be deployed in a manner consistent with both a robust edge pedestal and control of the steady heat flux to PFCs.

The high temperatures of PFCs that are envisioned in ARIES designs (500-1000^{o}C) to make the fusion power cycle more thermally efficient also bring new challenges and opportunities. The challenge is that the PFC materials and technologies to be used at such high temperatures are both at their heat load handling limits and have a narrow temperature operational window. The opportunity associated with high-temperature PFCs is to study hydrogenic retention and control the tritium fuel cycle; in a DEMO reactor less than 1 in 100,000 T ions striking the wall can be retained in the material. That should be contrasted with current devices where this ratio is found to typically be only ~1 in 100. The high hydrogenic diffusion and reaction rates at DEMO PFC temperatures will greatly aid removal of fuel ions from PFC surfaces and allow demonstration of the fuel control required for DEMO. The intense steady and transient loads on the PFCs, however, will result in dust production. Dust needs to be well-controlled for safety reasons in DEMO.

The flux of fusion neutrons also represents a great challenge for a DEMO reactor's PFC materials. While the neutron damage in the bulk PFC material may lead to increased tritium retention, the main impact of neutron irradiation is on thermal conductivity and brittleness, which affect the necessary thickness and structural integrity of the PFC components. For this reason the effects of neutrons are largely separable from the plasma-wall processes and resulting interaction with core plasma operation. Efforts to address the effect of neutrons on PFCs are encompassed in Thrusts 11 and 13.

Initial Analysis of the Machine Characteristics Required for this Thrust

The first task for designing a facility to support the central goal of this Thrust, demonstration of an integrated solution for the core and boundary, is to develop the requirements for such a facility's optimal scientific and technical capabilities. The first of these characteristics is the power density. There are two metrics that the community often uses for comparing the power density across devices (Table 1). Their different scalings reflect different heat flux issues and the large uncertainty in our present understanding. The first of these, P/S, is the average power through the surface of the plasma at its edge. The second metric, P/R, which assumes a constant width of heat flux in the boundary plasma, represents a more conservative projection to ITER and DEMO. A facility to support this Thrust should be designed for power flows into the edge and to the PFCs approaching DEMO levels (Table 1). Further work on heat flux width scaling, such as that now ongoing, and that proposed in Thrust 9, will lead to improved estimates of expected heat flux profiles that can be taken into account for specification of the heating power requirement.

To demonstrate that steady and transient heat flux control and other PFC solutions will be compatible with optimal core plasmas, the facility to support this Thrust must have the ability to obtain a robust edge pedestal. An important parameter for obtaining DEMO-like pedestal behavior is the edge bootstrap current profile that strongly affects edge stability and ELM dynamics. To approach DEMO conditions the edge collisionality must be low. Study of pedestal issues, as well as radiative mantle solutions, will add an additional requirement on the input power to be a significant multiple of the power required to obtain high confinement mode (H-mode). Taken together these capabilities will produce a robust pedestal with DEMO-like ELM characteristics as a test bed for the most promising techniques and technologies for controlling both steady-state and transient heat flux.

Plasma sustainment is also an important characteristic of the facility to support this Thrust in achieving very long pulses. The facility should be designed to allow for testing a variety of heating and current drive issues and options. Design of heating and current drive systems employing ion cyclotron resonance heating and lower-hybrid must include an examination of coupling efficiency through the expected dense boundary plasma and the strong pressure gradients in the pedestal into the core plasma. The launching structures must endure long periods of interaction with the boundary plasma. The design of neutral beam, electron cyclotron current drive and electron cyclotron heating systems will generally be less sensitive to edge issues, but here the focus will be on their capabilities for controlling the core over a sufficient range of densities, due to accessibility physics.

Another DEMO characteristic that will not be addressed by current or planned facilities is that of high temperatures for all components; this will be required in DEMO because higher operating temperatures lead to higher thermal conversion efficiency to electricity. The capability to attain such high temperatures (in the range 500 °C to 1000 °C), as well as explore their effects on core and boundary characteristics, is central to the goals of this Thrust. He coolant technology options are currently the most strongly considered for DEMO, but liquid metal coolants should also be considered. Tungsten PFCs will likely have an operational temperature window above the ductile to brittle transition temperature, but below the limit due to recrystallization. Liquid-surface PFCs are likely to have a lower maximum operating temperature due to their higher evaporation and sputtering rates at high temperature. Techniques to monitor and control dust production from solid PFC surfaces and evaporated material from liquid PFC surfaces will need to be qualified. Fuel retention and permeation into PFCs will also have a strong temperature dependence, and thus control of temperature will be needed to allow exploration of the effect on retention in PFCs and on the overall fuel cycle. Given the history of tokamaks showing a strong dependence of core energy and particle confinement on recycling and surface conditions, active control of surface temperatures could also be important for maximizing the performance of the core plasma.

While the ultimate goal for a steady-state DEMO is pulse lengths of order 10^7 sec with >80% duty factor, design and science trade-offs (e.g., costs of pulse length vs. gains in physics explored) should be considered in determining the appropriate specifications for a facility to support Thrust 12. In general, the requirement is that the cumulative particle and energy fluence to the plasma boundary and materials reach a level where the long-term evolution of the PFCs, due to erosion, fuel recycling, macroscopic PFC thickness changes (positive and negative) and the fuel residing in the PFCs, can be accurately extrapolated to DEMO. There is an extremely large range of potential time scales to examine, spanning from a few seconds for recycling equilibration (tungsten at 1000K), of order an hour for the fuel to equilibrate with the full PFC thickness (5 mm thick tungsten at 1000K), of order a day for erosion and redeposited thicknesses to reach 100 microns, and of order a month to erode through half the PFC thickness. Longer pulses with high duty factor will be highly advantageous, if not necessary, for studying critical physics and operational issues such as ELM and disruption avoidance, and dust production and removal. Relatively short pulses with high repetition rate may be the least costly to achieve, but the large number of thermal cycles required to study the high fluence issues may compromise the scientific and technical results. The determination of the optimal pulse length and duty factor for this device will require examination of the trade-offs of cost and reliability versus the issues to be studied.

The new data from this Thrust will be a primary resource for validation of the models that will be developed as part of Thrusts 9 and 10 that are needed for designing DEMO boundary components and operation. Model validation requires measurements to fully characterize the plasma and plasma-material interactions. This implies 2-D measurement of the plasma state, including density, temperature, power accounting and impurities. Measurements must also include full coverage of surface properties, including temperature, surface composition, erosion and redeposition rates, along with accounting of retained fuel. This requires extremely good diagnostic access, which must be part of the design from the start.

A key mission of this Thrust is validating the integrated performance of the material boundary with the core plasma in DEMO-like conditions. The output of Thrusts 2, 5, 9, 10 and 11 will be a number of promising PFC configurations, and control scenarios and actuators, that will have to be tested in an integrated fashion in this facility. It is therefore necessary that this facility feature flexibility for ease of multiple hardware changes driven by the broad testing program. In particular, this facility will undergo modifications in PFC technology and materials, boundary and divertor heat bearing configurations, actuator systems for control of the steady-state plasma and transients (ELMs and disruptions), and associated diagnostics to monitor all these systems. This flexibility is further needed to allow for unforeseen modifications, due to our present limited understanding of boundary and core plasma issues as well as the first-wall technology and their interaction. In summary, this facility should be capable of testing and validating the full variety of configuration options that can be applied to devices on the path to DEMO.

While complete boundary solutions must be employed and tested in a high-power D-T facility, the variety and number of issues to be addressed for the boundary and its interaction with the core plasma can be more readily examined in a limited activation environment. Such operation will have a number of advantages in carrying out this research. First, the lower levels of radiation will allow for greater access for flexibility to change out the variety of components, systems and configurations that must be tested. A limited activation environment would also provide a less harsh operating environment for diagnostics. Finally, the entire device scale and design will not be driven by the requirement to achieve high fusion power gain.

There are significant tradeoffs associated with the level and timing of deuterium operation, since activation by D-D neutrons will be a concern. While hydrogen operation would produce a minimum of nuclear activation and preserve manned access to the facility, a number of issues would benefit from deuterium operation. These issues include degraded core plasma confinement and less robust pedestal performance at given heating power with hydrogen, as well as isotopic scaling of scrape-off layer transport, sputtering and other plasma-material phenomena. Direct effects of neutron irradiation on plasma-material interactions, which are not anticipated to be strong, can nonetheless be examined in this facility by installing samples and/or components that have been previously irradiated. The balance and sequencing of hydrogen versus deuterium operation should be examined with respect to the trade-offs in flexibility, and requirements for shielding and remote maintenance, to maximize the scientific output of this facility.

Research Activities of the Facility Required for this Thrust

The large array of scientific issues to address in Thrust 12, as listed earlier, assures a rich and diverse scientific program. The Thrust provides a dedicated, enabling facility that will investigate the integration of an extremely wide-ranging set of core and edge scientific issues, while simultaneously implementing required technologies. Extensive diagnostics will be deployed to provide critical science information on the core and edge plasma, and on the evolution of PFC material surfaces.

A variety of steady-state heat flux solutions would be tested, which may include a radiative divertor and/or mantle, precision PFC alignment in conjunction with large flux expansion (e.g., vertical target, X-divertor, Snowflake) and expanded flux-tube divertors (e.g., Super-X). Solid and liquid PFC surfaces will be tested in accommodating DEMO-like heat flux, material temperatures and cooling technology. In both cases the solutions to be tested will be dependent on positive results in other thrusts. The ambient PFC temperature will be a particularly important parameter to vary in the effort to limit fuel retention, control the fuel cycle and potentially improve core plasma performance. Furthermore, the facility's long-pulse and diagnostic capabilities will for the first time allow a credible examination of the mechanisms controlling long-term material migration and PFC surface evolution in a DEMO-like environment.

Critically, these scientific and technological solutions to edge problems must be shown to be consistent with plasma sustainment and sufficient core plasma performance, such that steady-state operation is feasible within the window of allowable core plasma parameters. A controlled pedestal is central to this task since it the intermediary between the edge and core. A sufficient range of pedestal scenarios must be available and examined to *enable* the long-pulse plasma surface interaction science mission. Pedestal parameters, in particular particle density, will be varied to examine tradeoffs in confinement, bootstrap fraction and current drive efficiency toward simultaneously meeting the steady-state and edge missions. Core and edge integration will also require control of ELMs and perturbing disruptions, suggesting exploration of 3-D fields and innovative core confinement regimes. A battery of actuators will be used to drive current and modify the core profiles. All this must be demonstrated to have long-term reliability and be compatible with steady-state heat flux control while maintaining a robust pedestal.

Successful operation of this facility should demonstrate solutions for DEMO-like boundary conditions that are compatible with optimized core plasma operations. The timely development of these core-edge solutions should accelerate the successful operation of a Fusion Nuclear Science Facility and a DEMO power plant.

	EAST	JT-60SA	KSTAR	ITER	ARIES-RS	ARIES-AT
P_{in}/S (MW/m^2)	0.55	0.21	0.38	0.21	1.23	0.85
P_{in}/R (MW/m)	13	14	12	28	93	75
Pulse Length (sec)	400	100	300	300-3000	~ 10^7	~ 10^7
H Diffusion depth per pulse (m)	$6x10^{-6}$	$3x10^{-6}$	$5x10^{-6}$	$0.4\text{-}1x10^{-4}$	0.36	0.61
Annual $P_{in}t/S$ (GJ/m^2)	110[1]	30[2]	38[3]	210[4]	31,000	21,000
Species	D, H (D-D 10^5s/yr)[5]	H (D-D 10^4s/yr)[6]	H (D-D 2 10^4s/yr) [3]	D-T	D-T	D-T
Wall T (K)	300[7]	300	300	400	1000	1300
P_h/P_{LH}[8]	3.9	4.3	3.0	2.2	5.6	4.8

Table 1.

NOTES:

P_{in} = total input power, including alpha heating, for proposed scenarios. Bremsstralung and synchrotron losses are ~20 – 40% in ITER and ARIES and should be included in surface losses, but not divertor heat loading. Maximum planned external heating for non D-T devices.

[1] J. Li, private communication. Expected operation < 2 x 10^5sec/year.

[2] Y. Kamada, private communication: NB duty factor = 1/30. Operation assumed here = 12 hours/day, 100 days/year = 1.44 x 10^5 sec/year.

[3] 10^5 seconds per year total operation, $2x10^4$ seconds per year D-D operation.

[4] Assumed 2500x 400s shots per year.

[5] J. Li, private communication. 27 hours to allow entry into hall. Plan to assess possibility of remote maintenance.

[6] Yearly neutron budget/maximum neutron rate.

[7] J. Li, private communication. 500C W wall being assessed for EAST late operation phase.

[8] Martin, Y.R., FEC 2004, eq. 7, evaluated at $n = n_{G.}$ $P_h = P_{in}$ less bremstrahlung and synchrotron radiation, appropriate for use in evaluating $P/P_{LH.}$

Thrust 13: Establish the science and technology for fusion power extraction and tritium sustainability

As a practical energy source, a fusion power plant must create the tritium fuel it uses (by capturing fusion neutrons in the element lithium) and operate at high temperature so that the fusion energy can be converted efficiently to electrical power or other end uses. This Thrust aims to develop the scientific foundations of practical, safe and reliable processes and components that 1) harvest the heat produced by fusion, 2) create and extract the tritium from lithium, and 3) manage (radioactive) tritium that circulates in the plant. A continuous effort of experimental research and predictive model development, focused on the phenomena and interactions occurring in fusion nuclear components, is essential, both to prepare for next steps in burning plasma physics research as well as to accelerate progress toward practical fusion energy as the ultimate goal.

Key Issues:

- *How can fusion power be extracted from the complex structures surrounding the burning plasma? Can these structures be engineered to operate reliably in this extreme environment?*
- *How do we contain and efficiently process the tritium fuel in a practical system? Can this unprecedented amount of mobile tritium be accounted for accurately?*
- *How should lithium-bearing materials be integrated into power extraction components to generate tritium fuel to replace that burned in the plasma? Can simultaneous power extraction and fuel sustainability be achieved?*

Proposed Actions:

Perform fundamental research to establish the scientific parameters necessary to address the issues. An example activity is the exploration of tritium chemistry, heat transfer, and magnetic field interactions in lithium-bearing liquid metal coolants.

- Perform multiple-effect studies to understand the combined impact of the operating conditions and component complexity typical of a fusion environment. An example activity is utilizing the ITER burning plasma as a test environment to perform tritium breeding and power extraction experiments with relevant materials, instrumentation, component designs, and operating temperatures with necessary ancillary systems.
- Perform integrated experiments to characterize the complete effect of fusion conditions and facility performance. An example would be construction and operation of a Fusion Nuclear Science Facility (FNSF) to perform testing that resolves the remaining gaps stemming from the effects of significant surface heat flux and neutron irradiation over a long period of time in concert with all other fusion environmental conditions.
- Develop the accompanying theory and predictive models necessary to understand and apply the experimental results, and collect reliability, tritium accountability and safety data at all stages.

Introduction

Harnessing fusion power for useful purposes will require significant advances in the understanding of fusion fuel cycle and power extraction. ITER and later DEMO (an energy demonstration reactor) will be significant extensions beyond present experience (as described in the detailed Harnessing Fusion Power theme chapter), requiring the successful operation of systems that currently have never been fabricated, demonstrated, or tested in a relevant in-service environment. The primary focus of this Thrust is to address this situation by developing the science base and technological readiness for safe and reliable: 1) power handling and extraction 2) operation of the fusion fuel cycle, and 3) tritium breeding and extraction. The flows of power and tritium associated with this Thrust are shown on Figure 1:

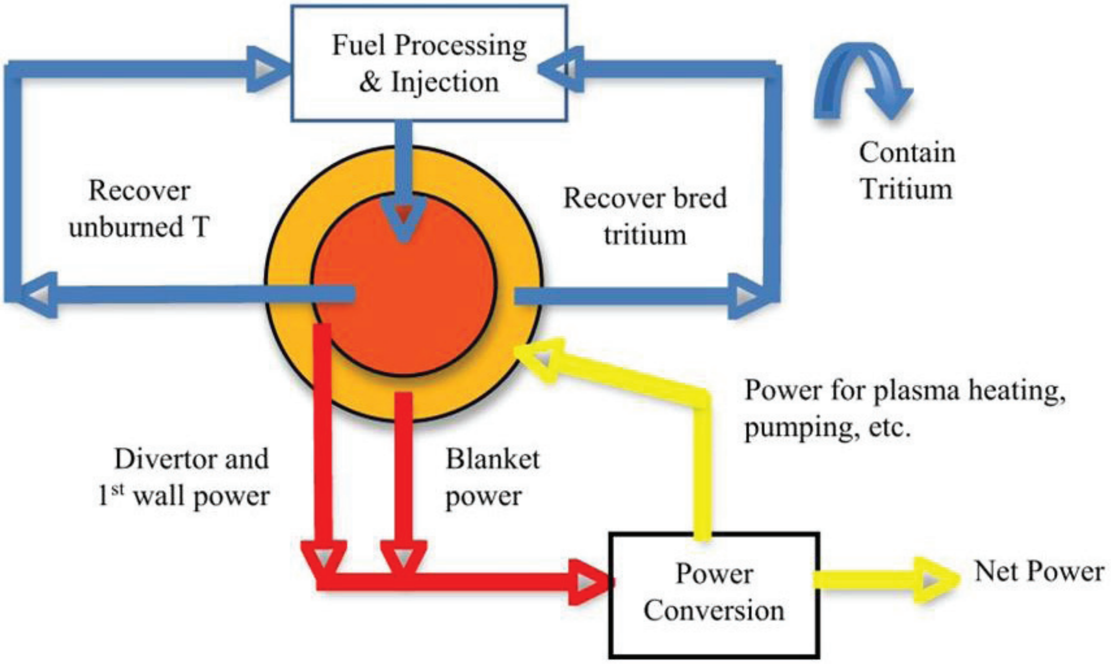

Figure 1. Schematic of power and tritium flow covered in Thrust 13.

Plasma Chamber ("In-vessel") Considerations — Tritium breeding and power extraction are key features of the plasma chamber surrounding the burning plasma, including the breeding blanket with integrated first wall, shield, and divertor. But the various components of the plasma chamber are subjected to an extreme fusion nuclear environment with many challenging conditions: (a) an intense flux of 14 MeV neutrons that access many high-energy threshold nuclear reactions to produce highly nonuniform nuclear heating, tritium, helium and other gases, atomic displacements, and many transmutation products; (b) intense fluxes of charged particles and radiation absorbed on surfaces exposed to the plasma; (c) strong magnetic fields with temporal and spatial variations; (d) electromagnetic and thermal coupling to the plasma including transient events like plasma disruptions and edge localized modes (ELMs); (e) high vacuum conditions; (f) high temperature operation; and (g) strong chemical activity. Understanding the behavior of components in this fusion nuclear environment represents a challenge requiring important advances in many scientific fields, engineering disciplines and technology development. This area is usually referred to as Fusion Nuclear Science and Technology (FNST). Thus, the in-vessel components are constrained primarily by a) survival in an extreme environment and b) the need for infrequent maintenance.

Ex-vessel Considerations — Tritium extraction from the in-vessel breeding zones, tritium processing of the plasma exhaust, and elements of power extraction associated with transporting hot coolant to the power conversion systems occur away from the plasma chamber. The primary ex-vessel constraints are: 1) radioactive hazards from tritium and other transmutation products hazards must be mitigated (e.g., exposure limits remain the same, while hazardous amount increases), 2) tritium inventory must be minimized, efficiently processed, and strictly accounted for, and 3) heat carried by gas or liquid metal coolants must be efficiently utilized for electricity production or process heat.

Guiding Principles — At each of the stages of development toward a demonstration of fusion energy (DEMO) there is a critical set of capabilities in FNST that needs to be in place to proceed further. Fusion development in general, including all aspects of plasma physics research, requires authoritative information on technology to evaluate technological readiness and identify paths toward a successful DEMO. While progress has been steady, new knowledge and increased effort is needed. Further progress can occur through an integrated program of well-instrumented benchmark and integrated experiments, first in labs, later in dedicated facilities and supported by validated computational models. Even in the research stage, an emphasis on pathways that improve the safety and reliability, and not just the performance, must be sought out and emphasized.

Proposed Actions

Approach — A host of gaps in sufficient understanding leading up to this Thrust have been identified and discussed in the Harnessing Fusion Power Theme Chapter. To optimally fill these gaps the following progression is proposed. First, fundamental research will be performed. Such experiments might not be prototypic (i.e., a test of "nearly final" component); rather this phase will focus on experiments designed to elucidate key, individual parameters and behaviors needed before more complex experiments can be successfully undertaken. This stage will be followed by experiments to understand multiple effects, e.g., more complex components, environmental conditions (heating, magnetic field, tritium, etc.) and subsequent synergistic phenomena. Finally, integrated tests will focus on the performance of a component in a fully representative operating environment, and/or of an overall facility. The overall progress is shown in Figure 2 and will be discussed in subsequent sections.

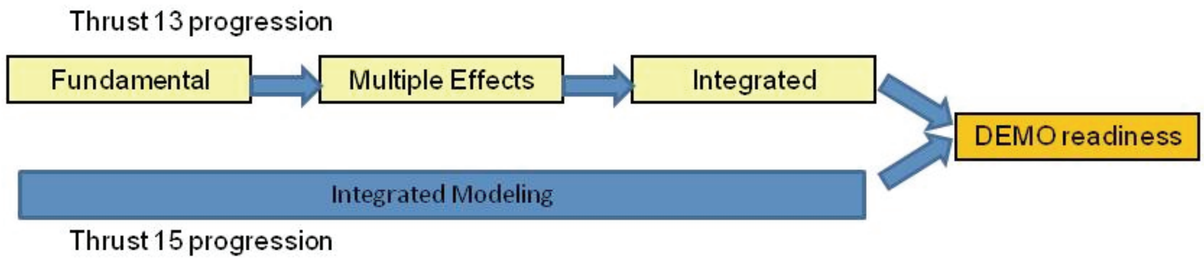

Figure 2. Progression of Thrust 13 activities from fundamental research to integrated tests. Supporting integrated modeling is discussed in the description of Thrust 15.

Each Thrust 13 element includes the related theory and modeling to help understand, apply and generalize the experimental results. These modeling efforts and codes are then available to Thrust 15, which integrates modeling results from a variety of areas, especially Thrust 14 on parallel material development, characterization and engineering.

This progression is similar to the concept of Technical Readiness Levels or TRLs used successfully in other fields to gauge and guide the progression of understanding and service readiness of complex technologies. This Thrust encompasses in its scope many different components needed for fusion power extraction and the tritium fuel cycle, and each of these components will have different concepts, materials, coolants, etc. So a broad program of research with concepts at different stages of readiness and development is foreseen. However, unification and direction within this Thrust will come from close coupling with Thrusts 14 and 15, Thrusts 11-12 concerning plasma facing components (PFCs) such as the divertor, and plasma-related thrusts emphasizing control, off-normal events and configuration.

Fundamental Research *— Perform fundamental research to establish the scientific parameters necessary to address power extraction and tritium fuel cycle issues.*

Advancing FNST knowledge at the fundamental scientific level is needed in many areas to complete the separate-effects database and establish the basic phenomenological and constitutive models necessary to understand tritium control, processing, and power extraction components for fusion. It is not possible to present a complete list, but example areas include:

- Lead-lithium alloy tritium chemistry, heat and mass transport characteristics, isotope and impurity control, etc.
- Liquid metal magnetohydrodynamic (MHD) interactions for liquid metal blankets, first-wall melt layers and free surface divertors.
- Tritium processing and containment (diffusivity, solubility, reaction rates, activity coefficients, etc.) for a variety of candidate materials and temperatures.
- Ceramic-breeder pebble-bed response to thermomechanical load and cycling.
- Interaction database of beryllium and liquid metal alloys with water and air.

Significant past work has been done in many of these areas, and progress has enabled the refinement and improvement of the conceptual design of components for fusion. However, gaps remain and an intensive program of laboratory-scale experiments and model development that address these gaps is envisioned. Emphasis should be placed on leading and alternative power extraction and fuel cycle systems, and on both safety and reliability aspects. The scope of effort in this first stage is roughly a handful of focused university and laboratory programs, and is not large compared to the current plasma physics program in the US. Enhancement in this area of research is essential.

Multiple-Effects — *Study interactions of multiple effects in operating components to understand the combined impact of the fusion environment and the complexity of typical fusion components.*

Power extraction and tritium breeding components (in-vessel) are geometrically complex, operate in a very complex environment, have multiple functions, multiple materials and many material interfaces. Tritium processing components (ex-vessel) must both produce on-spec products (relying on multiple fundamental parameters), and contain and account for tritium with high efficiency and accuracy. Synergistic phenomena will likely dominate the behavior, as well as failure modes and reliability of designs and prototypes.

Multiple effect experiments with different combinations of conditions are necessary, for instance, combined thermal and liquid metal MHD facilities, heat flux and magnetic field facilities, plasma-based facilities, fission reactors with integrated fusion coolant and breeder capabilities, etc. Such research will help (a) choose materials and designs that satisfy the competing requirements of the components, (b) provide data to verify models or identify areas in need of further fundamental research and model development, (c) compile the needed reliability and safety database necessary to validate codes, and (d) prepare for more fully integrated fusion environment testing in ITER or an FNSF, and for DEMO.

The testing facilities themselves can be upgrades of existing facilities when possible, and new facilities when required. Each facility and the program to build and perform the experiments can be thought of as roughly similar (but slightly smaller) scale to plasma physics “proof of principle” experiments. A significant subset of this effort should include the development and testing of engineering diagnostics and remote maintenance approaches, techniques and equipment. These capabilities are clearly needed for both for the design and operation of DEMO, but more immediately for performing fusion environment experiments and testing in future fusion devices. A specific program to develop these capabilities is urgently required, executed in conjunction with the experimental programs and facility designs that must use them.

One of the key upcoming multi-effect experimental opportunities will be the ITER Test Blanket Module (TBM) program. The ITER test module size, neutron flux, magnetic field and pulse length are each significant enough to provide a suitable test environment consisting of all important fusion conditions that affect first wall and breeding blanket systems and diagnostics. The strong, spatially complex magnetic field is a key parameter, especially for liquid metal blankets. The neutron damage, while generally small for structural materials, is significant enough to observe the impact of key phenomena in ceramic insulators or solid breeders, including swelling and electrical property changes. Determining how the combined environmental conditions and prototypic geometry of module-scale experiments affect performance will be invaluable information for first-wall blanket systems design, simulation validation, safety, licensing, and reliability growth programs. Significant prior separate effect and partially integrated experiments are a prerequisite, both to be able to fabricate and qualify TBM experiments for ITER, but also to be able to fully understand and interpret the experimental results.

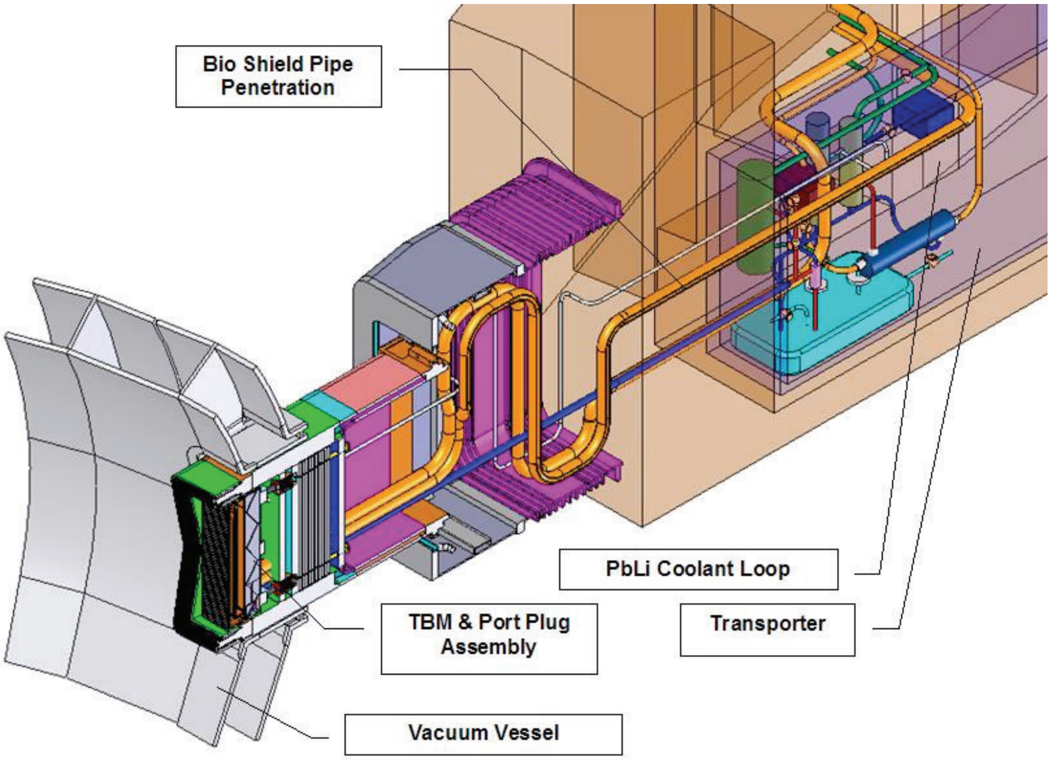

Figure 3. System concept for performing tritium breeding and power extraction experiments in ITER, using experimental modules with relevant materials, coolants, and support systems, at prototypic temperatures.

***Integrated Testing** — Perform integrated experiments to characterize component, multiple component, and facility performance in a fully representative operating environment.*

Previous experimental work has provided the groundwork, but will not replace the need for integrated testing as the last stage to resolve key knowledge gaps for power extraction and fuel cycle components and facilities. This is especially true for in-vessel components where a Fusion Nuclear Science Facility (a.k.a. Component Test Facility, Fusion Development Facility and Volumetric Neutron Source) will be required to fully establish the engineering feasibility, reliability, safety, fuel sustainability, and performance verification of the plasma chamber components. Such a facility would provide a test environment with significant neutron flux and fluence, in concert with all other fusion conditions such as long operating times, magnetic field, etc.). The program should include both highly instrumented scientific experiments to test, discover, analyze, and understand the integrated phenomena and behavior, as well as aggressive iterations using multiple modules for reliability statistics. It is also possible to use the facility to research, develop, and demonstrate tritium self-sufficiency of a complete fuel cycle.

Integrated testing will cover:

1. Multiple, interacting effects in components activated only in a full fusion environment.
2. Interactions among tritium breeding, extraction, processing, and containment in the presence of plasma-wall interactions (PWI), and the plasma facing component interactions with in-vessel and ex-vessel components.

3. Buildup of tritium and impurity concentrations, and distribution to equilibrated levels.
4. Impact on component behavior by radiation damage of in-service materials.

Perhaps this can be done in a single staged facility or perhaps more than one facility will be required. The scale of effort will be large, and the planning and design of such a nuclear facility is itself an intensive task. The mission, capabilities, and time scales for such facilities must be carefully considered in a DEMO readiness program.

Connection to Safety and Reliability — *Develop theory and predictive models, and collect reliability and safety data at all stages.*

Incorporation of activities in Thrust 13 will be instrumental in the design of an FNSF and subsequent DEMO, which in turn will need to meet demanding safety and Reliability, Availability, Maintainability, and Inspectability (RAMI) requirements described in the Harnessing Fusion Power Theme Chapter. While the research activities in Thrust 13 stem from establishing the feasibility of fusion power extraction and the tritium fuel cycle, they will naturally result in opportunities to collect and document information on RAMI and safety. As components begin to fail during testing, the conditions and mode of failure will be identified, understood, and documented. Improved test conditions and components can then be designed and introduced for further testing to build a critical RAMI database. For safety, fundamental constitutive behavior and reaction rates from complex systems behavior and failure modes will form the database for safety source terms and modeling. Experiments needed to challenge hazards mitigation will be identified and performed as part of the testing program.

Connections to Other Thrusts and Themes in Plasma Science and Engineering

In addition to the close linkage to Thrusts 14 and 15 (already discussed), research on fusion power extraction and the tritium fuel cycle must be closely linked to other thrusts, notably those focused on PFCs, PWI, internal components, plasma configuration, and burning plasma.

The first wall, divertor power extraction and tritium control are intimately connected to PFC and PWI issues. Operation of plasma contact surfaces at high temperatures and with neutron damage is likely to change their behavior vis-à-vis tritium retention, recycling, diffusion and impurity generation. The potential use of liquid metal free surface divertors, or the occurrence of melt layers during off-normal plasma events, also requires concurrent understanding of the MHD flow dynamics and control, coupling between liquid surfaces and the plasma, as well as the impact on edge behavior, particle pumping, tritium burn fraction and plasma impurity control. Thrust 13 attempts to be inclusive of power extraction and tritium fuel cycle issues critical for the successful operation of PFCs. However, Thrusts 11 and 12 also include a detailed strategy to address many of these gaps in concert with plasma wall material development and interactions in special test facilities. The research described in Thrusts 11 and 12 should be in concert with the test facilities described as multiple effects testing in Thrust 13, with emphasis on coupled power extraction, tritium and plasma material interaction issues.

The plasma magnetic configuration, operational modes, and the plasma control hardware (most notably Thrusts 1, 5, 16, 17, 18) will have a strong impact on establishing tritium self-sufficiency requirements. Understanding the fueling efficiency, tritium burn fraction, the size and materials used in antennae, coils, shells needed for plasma heating and control, and many other factors must be considered jointly between the plasma requirements, materials and technological limitations and the impact on tritium fuel sustainability. In a similar fashion, leading candidate structural materials for power extraction and tritium fuel cycle components are based on ferritic/martensitic steels that have nonunity relative magnetic permeability. Error field effects from the use of these materials in the plasma chamber, as well as from MHD currents arising from flowing liquid metal alloy coolants, must be jointly assessed from plasma and technological perspectives. Many of magnetic fusion's most difficult problems: disruption, VDEs, ELMs prediction, control, mitigation and survivability (Thrust 2); concurrent steady-state current drive (Thrust 5) and power extraction; significant radiation damage in in-vessel components; tritium fuel sustainability, etc., require integrated physics and technology research and solutions — and test facilities in which to pursue them. The idea of pursuing a Fusion Nuclear Science Facility in which these problems can be investigated and resolved prior to a DEMO is a unifying theme in many of the thrusts. The effort to establish the requirements and the design of such a facility in the nearer term will require participation by many if not all of the other thrusts.

Conclusion

Developing the needed power extraction and fuel cycle components, with the necessary performance, reliability, maintainability, and safety characteristics for fusion, is a critical prerequisite to designing nuclear plasma chamber systems for any future burning plasma devices and ultimately an energy producing DEMO. There has been past significant work in the US in a number of critical research and design areas, and US design concepts are widely used and respected internationally. But the general field of Fusion Nuclear Science and Technology has not been supported adequately in the US over the past decades and there has been a worrisome loss of capabilities and human resources. The US still has a core set of capabilities and niche areas of expertise, but we must immediately expand efforts in FNST basic property and separate/multiple effects experiments and simulation in a number of needed areas. An investment in human, facility, and computational infrastructure will help reinvigorate FNST, and will also provide the tools to improve American competitiveness in fusion where the US has traditionally been a scientific leader and innovator. In addition, basic FNST research in fluid, thermal, and nuclear sciences; electromagnetics; material science; hydrogen chemistry in metals and ceramics; granular media thermomechanical behavior; novel measurement techniques; high-performance computing; and many other fields can lead to innovative methodologies, computational tools, materials and technologies with relevance to a wide variety of energy, chemical, nano/bio, computational science, and industrial applications. Excellence in Fusion Nuclear Science and Technology is a prime example of the spirit of the American Competitiveness Initiative because of its central importance to the development and demonstration of inexhaustible fusion energy, and its wide application in other strategic areas.

Thrust 14: Develop the material science and technology needed to harness fusion power

Fusion materials and structures must function for a long time in a uniquely hostile environment that includes combinations of high temperatures, reactive chemicals, high stresses, and intense damaging radiation. Ultimately, we need to establish the feasibility of designing, constructing and operating a fusion power plant with materials and components that meet demanding objectives for safety, performance and minimal environment impact.

Key Issues:

- *What thermal, mechanical, and electrical properties are needed to meet fusion objectives?*
- *How does radiation damage affect the properties of materials?*
- *How do synergistic effects involving radiation damage, high temperatures, high stresses and corrosion phenomena affect the feasibility of operating a fusion power plant?*

Proposed Actions:

- Improve the performance of existing and near-term materials, while also developing the next generation of high-performance materials with revolutionary properties.

  Understand the relationship between material strength, ductility and resistance to cracking.

  Design materials with exceptional stability, high-temperature strength and radiation damage tolerance.

  Understand how interactions with the plasma affect materials selection and design.

  Establish the scientific basis to control the corrosion of materials exposed to aggressive environments.

  Develop the technologies for large-scale fabrication and joining.

  Determine the underlying scientific principles to guide discovery of revolutionary high-performance materials while minimizing radioactive waste and maximizing recycling.

- Develop and experimentally validate predictive models describing the behavior and lifetimes of materials in the fusion environment.
- Establish a fusion-relevant neutron source to enable accelerated evaluations of the effects of radiation-induced damage to materials.
- Implement an integrated design and testing approach for developing materials, components, and structures for fusion power plants.

- Use a combination of existing and new nonnuclear and nuclear test facilities to validate predictive models and determine the performance limits of materials, components and structures.

The purpose of this Thrust is to provide the fundamental materials science and technology necessary to enable design, construction and operation of a fusion power plant. The activities of this Thrust must be fully integrated with those of Thrusts 9, 10, 11, 12, 13 and 15. In this Thrust the basic materials property information and models of materials behavior in the harsh fusion environment will be developed. This information provides the foundation for selecting, developing and qualifying materials to meet design requirements. Thrusts 10, 11, 12 and 13 will carry out progressively more integrated tests that use the materials and models developed in this Thrust. This is a highly iterative process since evaluation of materials performance from integrated tests will be used to refine materials selection and processing pathways, which in turn determines the available design window.

Introduction

For fusion to find its way into the energy marketplace it must compete economically with other energy options, and it must be developed as a safe and environmentally acceptable energy source, particularly from the viewpoint of radioactivity. Achieving acceptable performance for a fusion power system in the areas of economics, safety and environmental acceptability is critically dependent on performance, reliability and lifetime of the first-wall, blanket and divertor systems, which are the primary heat recovery, plasma purification, and tritium breeding systems. Design and performance of these key components are in turn critically dependent on the properties and characteristics of the structural materials. Since these are primary fusion power feasibility issues and since resources are limited, the main focus of recent fusion materials research has been on structural materials. Based on safety, waste disposal and performance considerations, the selection of structural materials for blanket and divertor applications is largely limited to reduced activation ferritic/martensitic (RAF/M) steels, dispersion strengthened ferritic alloys, tungsten alloys and silicon carbide composites. However, it is fully recognized that a host of other materials, such as plasma facing, breeding, shielding, insulating, superconducting and diagnostic materials, must also be successfully developed for fusion to be a technologically viable power source. A comprehensive research program must address the specific materials science issues associated with each of these materials systems. This Thrust provides the principal source of knowledge on fundamental materials behavior and radiation effects to support development of radiation-resistant materials needed in Thrusts 1, 7, 9, 10, 11, 12, and 13.

Description of Thrust Elements

Improve the performance of existing and near-term materials, while also developing the next generation of high-performance materials with revolutionary properties.

Temperature window and severe lifetime limits imposed by the in-service degradation of materials properties are the major challenge in the quest for fusion power. The major in-vessel systems will have a finite lifetime and will require remote maintenance and replacement. High reliability over the entire plant lifecycle is key to favorable economics and requires protecting against frequent minor failures and the imposition of severe operational limits. Along with lifetime, reliability is primarily determined by the performance of materials and components. Materials and structures must also provide acceptable safety margins during both normal operation and off-normal events. Radioactive isotope inventory and release paths are key considerations in designing for safety. The initial levels of radioactivity of materials on removal from service, and the rate of decay of the various radioactive isotopes, dictate acceptable storage and disposal methods and the possibility for recycle or clearing of materials. These issues are major considerations in the environmental acceptability of fusion.

A significant expansion of the current fusion materials research effort is needed to fully explore the performance limits of first-generation power plant materials such as RAF/M steels (including dispersion strengthened ferritic alloys) and tungsten alloys, and to discover the next generation of high-performance materials. To achieve the widest possible operating temperature window, the underlying physical mechanisms controlling properties of materials at low, intermediate and high-temperatures must be determined. A key technical issue is the fundamental relationship between material strength and ductility or fracture toughness. Present engineering materials do not exhibit simultaneously high strength and high ductility or fracture toughness. Enormous technological benefits will be accrued far beyond fusion energy if this puzzle can be solved. Its solution will require advances in understanding how materials with high-defect densities deform, as well as the conditions under which cracks propagate, and in determining the role He and H make to hardening and non-hardening embrittlement of materials.

Another significant issue that has broad applicability beyond fusion energy is microstructural stability and deformation behavior of materials at high temperatures. Severe time-dependent, thermo-mechanical, high-temperature loading of large and complex fusion structures may be a grand challenge in itself, not considering radiation damage effects, representing a regime far beyond the limits of present-day technology. A major scientific challenge is to develop physical models of high-temperature creep and creep-fatigue interaction mechanisms. Current treatments of these phenomena are largely empirical and material-condition specific. High-temperature design rules evolved from a body of structural application experience that is enormously less complex, with fewer interrelated parameters, than in the case of fusion. For example, to move beyond the current state-of-the-art will require significant advances in models of creep deformation. These models must simultaneously treat the evolution of complex dislocation, interface and precipitate structures, stress-driven dislocation motion, the interaction of dislocations with obstacles, sliding of grain boundaries and diffusionally accommodated creep that accounts for multiple effects of grain boundary precipitates. The challenge of high-temperature performance will be enormously amplified by the effects of high concentrations of He, which can degrade performance sustaining properties, like creep rupture life, by many orders of magnitude.

In addition, plasma facing components (PFCs) experience severe and variable surface heat loads, damage from energetic ions, and erosion of material by ions (or neutrals) or by redeposited material from the plasma. These conditions will further complicate their microstructural evolution and may induce modes of failure that differ from materials elsewhere in the system. Furthermore, impurities from erosion directly affect plasma stability. Understanding how plasma-surface interactions affect the design and selection of PFCs and other in-vessel materials is critical. This area is an interdisciplinary specialization where some aspects of fundamental materials modeling, as well as experiments, are shared with Thrusts 9, 10 and 11.

Chemical interactions between the structural materials and other materials in a power system, such as corrosion of the structural materials by the coolant, oxidation in gaseous environments, mass transfer within the system via the coolant, and interaction with tritium breeders, must be determined and understood. Also, the impact of such phenomena on design and performance of the fusion power system must be assessed. Determinations of what chemical interactions are potentially important, understanding the chemical thermodynamics and kinetics, and providing the basic information for design studies are needed. This research requires specialized facilities such as high-temperature gas reaction systems for oxidation studies, thermal convection and small pumped loops to investigate corrosion and mass transfer phenomena with liquid metal coolants and, in some instances, specialized mechanical test systems to investigate dynamic effects of chemical interactions on mechanical properties. The selection and development of materials must be carefully integrated with efforts in Fusion Nuclear Science and Technology (Theme 3 and Thrust 13) through which the program will address the complex and often competing performance requirements for materials.

Central to successful deployment of fusion power systems is development of large-scale fabrication and joining technologies. A promising new alloy class, known as nano-structured (dispersion strengthened) ferritic alloys has recently been developed. These materials contain an ultrahigh density of nanometer-scale, non-equilibrium, enriched phases that may impart unprecedented high-temperature creep strength, radiation damage resistance, and efficiently trap He. However, large-scale fabrication and joining technologies that would enable construction of large geometrically complex components and structures from these materials do not exist. To exploit these promising materials, development of such technologies is essential. A partnership with industry to take advantage of large-scale fabrication and joining capabilities and experience is needed to make progress in this area.

Finally, a key challenge is development of high-performance materials that will ensure the economic attractiveness of fusion power plants while simultaneously achieving safety and environmental acceptability goals. Radioactive isotope inventory and release paths are important considerations in designing for safety. Development of low or reduced activation materials is central to ensuring that materials removed from service are recyclable and/or clearable, and will not require long-term geological disposal, thereby minimizing the impact on the environment. This will be primarily accomplished by reduction of impurities that restrict the free release of in-vessel components.

Develop and experimentally validate predictive models describing the behavior and lifetimes of materials in the fusion environment.

New materials must be discovered to make fusion a technically viable and economically attractive future energy source. The most efficient approach to materials discovery is a science-based effort that closely couples development of physics-based, predictive models of materials behavior with key experiments to validate the models. Materials degradation in the fusion neutron environment is an inherently multi-physics, multi-scale phenomenon. Models describing neutron irradiation damage processes in fission and fusion nuclear systems have been under development for many years. Figure 1 shows molecular dynamics simulations demonstrating that high-energy fu-

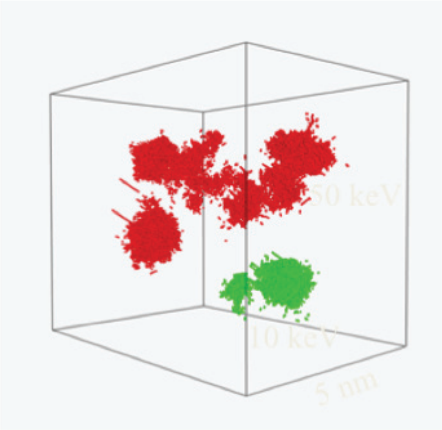

Figure 1. Molecular dynamics simulations demonstrating that high-energy fusion damage events (red atoms) are similar to multiple, lower-energy events (green atoms), which permits fission-fusion damage scaling.

sion damage events (red atoms) are similar to multiple, lower-energy events (green atoms), which permits fission-fusion damage scaling. However, the fusion irradiation environment is very complex due to displacement damage from high-energy neutrons, bombardment of the surface by energetic ions and concomitant high concentrations of damaging He and H from nuclear transmutation reactions, as well as surface modification by interactions with the plasma. Beyond the fusion nuclear environment, the materials, components and structures of a power system will be exposed to severe thermo-mechanical loads and aggressive chemical species. The current fusion materials theory, modeling and simulation effort needs to be significantly expanded at all spatial and temporal scales to address the following needs:

- Computationally efficient and physically robust interatomic alloy potentials, which account for directional bonding, magnetism and charge transfer (to accurately describe complex multi-component, multi-phase materials).
- Advanced large-scale, atomistic models that describe the very large number of material parameters and processes that interact in complex ways to control the migration, interaction, and accumulation of defects and gases, as well as the non-equilibrium rearrangements of solute constituents by segregation and phase transitions (to predict nanoscale evolutions in complex materials for both processing and extended service).
- Linked atomistic, mesoscopic, and continuum deformation and fracture models (to predict hardening, plastic instabilities, transitions from ductile-to-brittle and creep/creep rupture behavior for complex materials and loading conditions).
- Large-scale structural models that integrate all degradation phenomena (needed for virtual integrated testing and materials-component development).

Establish a fusion-relevant neutron source to enable accelerated evaluations of the effects of radiation-induced damage to materials.

To become economically viable, in-vessel fusion power systems will require structural materials with lifetimes approaching 200 displacements per atom (dpa). Recent fusion materials research and development efforts have led to the development of RAF/M steels with good resistance to irradiation in fission reactors for doses up to about 30 dpa. However, their performance in a high-

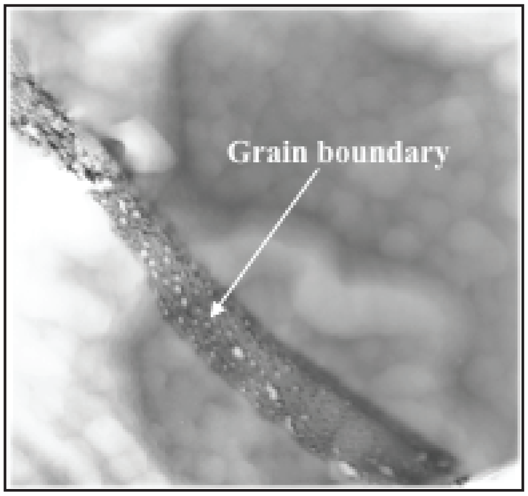

Figure 2. He bubbles on grain boundaries can cause severe embrittlement at high temperatures.

dose fusion-relevant environment with accompanying high He and H generation from nuclear transmutations is unknown. Because He substantially affects bulk properties, it is essential to carefully quantify its effect in order to develop a safe and reliable fusion power system. The figure above illustrates how He bubbles on grain boundaries can cause severe embrittlement at high temperatures. Therefore, a fusion-like neutron source is absolutely essential to generate a bulk materials property irradiation database necessary for design, construction, licensing, and safe operation of next-step fusion nuclear devices and fusion power plants.

Exploring the simultaneous effects of neutron displacement damage and He production by transmutations on a wide range of materials requires a fusion relevant neutron source. This is the conclusion of various high-level international assessments. The number of He atoms produced divided by the number of atoms displaced (He/dpa ratio) is an important fusion relevant attribute. The need to generate a database of environment-specific material properties is consistent with experience in all previous nuclear and other safety-sensitive nonnuclear technologies. There are several options for creating fusion-relevant neutron irradiation conditions, such as the proposed International Fusion Materials Irradiation Facility, the Materials Test Station, and the Dynamic Trap Neutron Source. Consequently, a key activity in this Thrust element is to carefully evaluate the various options and select the most technically attractive and cost effective approach or combination of approaches. Key criteria for a fusion relevant neutron source include flux, test volume and operational availability sufficient for accelerated testing to end-of-lifetime doses in a reasonable time. More specifically ≥0.5 liter volume with ≥2 MW/m^2 equivalent 14 MeV neutron flux to enable accelerated testing up to at least 10 MWy/m^2 (with larger volumes at lower neutron fluences for testing larger components), availability ≥70%, and flux gradients ≤20%/cm. These criteria will form the basis to evaluate the various options for conducting materials irradiation studies. Selections will be made that balance the need to obtain relevant bulk material property information with the cost, schedule and potential for international participation to leverage investments by the US.

A later possibility might be to include a provision for materials irradiation capabilities as part of a large-scale nuclear facility such as the proposed Fusion Nuclear Science Facility. However, it must be emphasized that bulk material property data from a fusion relevant neutron source would inform the design, construction and licensing of such facilities. Further, preceding the development and operation of any nuclear facilities, many nonnuclear test facilities will be needed to qualify materials for the unprecedented thermo-mechanical loading conditions that will occur in next-step fusion nuclear devices. Finally, multi-physics models of time-dependent structural performance and lifetimes must be established, so that benchmarking in the nonnuclear and nuclear facilities can be translated into a meaningful basis to design the next generation of fusion reactors.

Implement an integrated design and testing approach for developing materials, components, and structures for fusion power plants.

Design of fusion power systems faces a significant challenge in that neither the materials property database, nor the requisite computational tools, nor the fusion-relevant design rules and codes currently exist for reliable integrity and lifetime assessments of fusion reactor components and structures. Existing thermo-mechanical property data and high-temperature design methodologies are not adequate to permit even preliminary designs of a fusion power system. Development of fusion relevant design rules and tools are needed. Successful design of a fusion power system will require close integration between materials development activities and system design processes. The commonly used “function-oriented” material design process must be replaced with a concurrent material-component-structure design process.

Fusion energy clearly presents an enormous materials-structural engineering challenge given the unprecedented requirements of an attractive power system. For example, new design and in-service performance computational tools must be developed to replace simplistic high-temperature design and operational rules. These tools must ultimately be incorporated in design codes and regulatory requirements. Neither the designer nor the material developer can proceed without input from the other, and the development of materials requires and becomes an integral part of the system design process.

Material models, structural models, and design codes must be combined with models of damage and history-dependent synergistic failure paths that are controlled by complex interactions of numerous variables, processes and properties in the fusion environment. Guided by engineering design information, the integrated models must be informed by well-designed experiments, supported by high-quality material property databases that underpin models of the effects of long-term service, and benchmarked by pertinent component-structure level testing.

The required program includes a significantly increased commitment to materials-oriented efforts in how materials designers, engineers and developers can develop and incorporate the tools necessary to move forward. It also includes significantly increased commitments, and complementary and strongly coupled efforts, in the development of Fusion Nuclear Science and Technology (Thrust 13), in plasma facing components and plasma-surface interactions (Thrusts 10 and 12), and overall integration (Thrust 12 and others). The activities of this Thrust element will also be closely coordinated and linked with Thrust 15.

Use a combination of existing and new nonnuclear and nuclear test facilities to validate predictive models and determine the performance limits of materials, components and structures.
Full qualification of the materials, components and structures of any future fusion power system will require dedicated nonnuclear and nuclear testing facilities to allow system level testing of components, such as blanket modules and plasma facing components. This testing will look for synergistic effects and failure paths that are not revealed in single-variable material property studies. Tests of this type will help to validate and benchmark models of materials and structural performance, and provide insight on possible synergistic phenomena that could compromise system reliability and performance. Such experiments must be fully informed by the material property models derived from the separate-effects studies in a prototypic fusion environment and structural models validated by semi-scale type experiments. Discovery and development of new materials will be the primary objectives of this Thrust. The testing and evaluation of materials, components and structures under simulated fusion conditions involving progressively more integrated tests will be carried out in Thrusts 11, 12 and 13.

Scale and Readiness

The scale of this Thrust is very large. While specific cost estimates were beyond the scope of the present planning activity, an order of magnitude level of effort can be obtained by examining previous cost estimates prepared for a similar planning effort. The 2003 Fusion Energy Sciences Advisory Committee Development Plan estimated about $1,600M would be required for materials research and development over 35 years (not inflation adjusted) and would involve approximately 60 full-time scientific and technical staff at peak activity.

It is also instructive to look at efforts to develop high-performance materials for other demanding applications. For example, a worldwide effort involving hundreds of scientists over a period of 20 years overcame several major technical hurdles to successfully develop Ni_3Al intermetallic high-temperature alloys with sufficient ductility and fabricability for commercial applications in 2003. Similarly, a dedicated science-based 14-year, $175M effort recently developed four successfully improved generations of Si_3N_4 ceramics with a resultant four-fold increase in high-temperature strength. This development enabled monolithic ceramics to become reliable structural materials for certain low-stress applications. The historical precedent for development of structural materials ranging from aircraft turbine components to pressure vessel materials is that a period of 10-20 years with $150-200M budgets are required for successful development. The development challenges for these materials systems pale by comparison to that for fusion materials, which could be the greatest materials development challenge in history. The combination of high temperatures, high radiation damage levels, intense production of transmutant elements (in particular, He and H), high thermo-mechanical loads that produce significant primary and secondary stresses, and time-dependent strains for which no high-temperature design codes exist requires very high-performance materials. Consequently, it is imperative to initiate a robust materials science program now so that the needed information will be available at the appropriate time.

Other Scientific Benefits

Many of the materials science questions facing fusion energy are universal in scope and fundamental in nature with potential broad applications in fields beyond fusion energy. As mentioned, the thermo-mechanical demands being placed on fusion materials are at the limits of today's materials science and technology, even without considering the additional demands created by radiation damage. For example, the scientific challenges associated with developing high-strength, high-temperature materials that are micro-structurally stable are faced in many technologically demanding industries such as aerospace, transportation, and conventional power generation. Similarly, discovery of the means to simultaneously achieve high-strength and high-ductility or fracture toughness would represent an enormous breakthrough with broad implications for extending the lifetimes of materials that function in severe thermo-mechanical environments. In addition, fusion will develop recycling techniques and guidelines for the free release of materials, both of which are beneficial to the existing nuclear power industry and regulatory agencies.

Thrust 15: Create integrated designs and models for attractive fusion power systems

Capturing the energy released by a burning plasma and converting it to a useful power flow in a safe, reliable, and sustainable manner requires the successful integration of many systems and physical processes. This Thrust includes two primary aspects: 1) conducting advanced design studies; and 2) developing integrated predictive models.

Detailed advanced design studies bring to light key design trade-offs and constraints within an integrated system for DEMO. For example, maximizing power cycle thermal efficiency requires operation at high temperatures, with upper limits set by coolant and structure compatibility, thermal stress, material behavior and plasma thermal loading considerations. Identifying serious limiting factors and design issues can guide research toward high-leverage and high-payoff issues while minimizing program risk. Such detailed advanced design studies also allow the assessment of integrated safety, environmental and RAMI (reliability, availability, maintainability and inspectability) issues for fusion.

Advanced design activities are also essential in evaluating alternatives for a Fusion Nuclear Science Facility (FNSF), which is an important and necessary step on the path to fusion energy.

Integrated models are used to help reveal important science and technology interrelationships and interpret the results of fusion experiments and component tests, thereby reducing the risks and costs in the development of fusion nuclear systems. They will also support the advanced design studies for future fusion facilities.

Key Issues:

- *What advanced design studies are needed to identify system integration issues; optimize facility configuration; extend the operating parameter space for future fusion facilities that meet availability, maintainability, safety and environmental goals on the path to fusion power; and guide the R&D on high-leverage, high-payoff issues?*

- *How can separate physics, nuclear, and engineering models be most effectively coupled to address highly integrated fusion system behavior?*

Proposed Actions:

- Determine and improve essential aspects of fusion energy through advanced design and integration activities for DEMO, including:

  Optimizing the integrated configuration and maintenance approach to minimize risk and achieve the availability, maintainability, safety and environmental requirements for DEMO to be attractive.

  Determining the scientific basis for sustainable fusion power and identifying research efforts to close the knowledge gap to DEMO.

Evaluate, through advanced design and system studies, alternative configurations and designs for FNSF and the extent to which they address the research requirements.

Develop predictive modeling capability for nuclear components and associated systems that are science-based, well-coupled, and validated by experiments and data collection.

Extend models to cover synergistic physical phenomena for prediction and interpretation of integrated tests (e.g., ITER Test Blanket Module — TBM) and for optimization of systems.

Develop methodologies to integrate with plasma models to jointly supply key first wall and divertor temperature and stress levels, electromagnetic responses, surface erosion, etc.

Motivation and Background

A broad-based effort is needed to understand, model and evaluate the various components, subsystems and processes that must be integrated to create an acceptable fusion power plant (including closing the tritium fuel cycle and efficiently harnessing the fusion power). This will provide understanding of very complex interrelated behavior of systems that is otherwise difficult to anticipate, and could dramatically reduce the cost and risk of future R&D. Thrust 13 and 14 research focuses on the fusion fuel cycle, power extraction and materials, and modeling tools that will be developed and used to investigate specific aspects supporting this R&D. Thrust 15 focuses on integration; it includes conducting integrated advanced design studies focused primarily on DEMO, but also on nearer term fusion nuclear facilities, and integrating simulation models. The integrated models will simulate increasingly complex, coupled aspects of the fusion plant with a focus on the plasma chamber, breeding blanket, fuel cycle and power extraction.

Integrated Advanced Design Studies

1. Integrated concept development and system studies. A key element of this Thrust is an advanced integrated design effort for next-step experimental fusion nuclear facilities and DEMO. An FNSF has been proposed as a next-step device to fully establish the engineering feasibility, reliability, safety, fuel sustainability, and performance verification of the plasma chamber components with significant neutron flux and fluence, long continuous operating time, and with the buildup of transmutation products in prototypic materials. The first stage of an FNSF effort should include: defining its mission and requirements, and conducting conceptual design and system studies of different options to better assess their expected performance and capability and the extent to which they address the research requirements.

Design and system studies, such as those performed as part of the ARIES program, have proven extremely valuable in charting the path to fusion power and guiding research toward high leverage, high pay-off activities. As we move forward, we see this evolving into much more detailed design activities for DEMO with a substantially larger scale effort. This will enable a thorough understanding of the integrated performance (including RAMI and safety) of fusion systems, which will ultimately lead to procuring components and building facilities, similar to the detailed designs for ITER. Other countries, such as the EU and Japan, are also pursuing efforts on DEMO and this activity will include coordination with them (for example, through organized workshops).

Developing designs and systems models for all aspects of the plant, from plasma control to net power production, as an integrated team effort is essential to reveal key trade-off results and constraints, and to identify the design space parameters leading to the most attractive possible design. This integrated advanced design activity was identified as a key component needed to advance the areas of RAMI and safety and environment because it is only in the context of detailed designs that these aspects of fusion power can be fully evaluated.

Figure 1 illustrates some of the key fusion fuel cycle and power extraction processes that must be efficiently integrated into a working power plant. A comprehensive design study would include many other physical systems such as shielding, coils, power supplies, structures, etc. Table 1 lists aspects of the plant that must be evaluated in an integrated design study, including those related to necessary features for attractive fusion power (good economics, high availability, safe, low environmental impact).

2. Addressing RAMI research needs. Achieving high availability on DEMO requires an integrated design that promotes reliability and maintainability. High-level design choices will have to be made taking into account reliability and maintainability impacts. A maintainable configuration for the fusion core must be developed, perhaps using large, integrated in-vessel component modules that are time efficient for remote exchange between the reactor and hot cell, with off-line refurbishment performed in the hot cell. Standardization of components, in-service monitoring of equipment health, reliability and maintainability-centered design, component and subsystem redundancy, and fault tolerance also promote high availability. The underlying problem is that any power core in-vessel failure, regardless of how small, that causes the plasma to shut down and requires an in-vessel intervention to repair, will initiate a shutdown, repair, and startup sequence that is disastrous to plant availability. The DEMO integrated design activity must address the remote handling systems and hot cell facilities. Fusion facilities (such as FNSF), which are proposed to test reactor prototypical in-vessel components, also provide a needed opportunity to develop and test remotely maintainable in-vessel components using reactor remote handling systems and hot cell facilities.

The integrated design activity would provide needed guidance for fusion development:

- Studies would be performed to determine the R&D activities and the cost and schedule required to meet the DEMO availability goal, building on the scientific basis for component reliability engineering. Results would be used to plan development and test activities.

- Trade studies would be conducted to evaluate configuration options and component design alternatives, and would select those options that promote performance, availability, safety, environmental and cost objectives. Results would be fed back to supporting experiments and test activities to provide needed focus.

- A clear map for coordinating design integration and development and testing activities, and downselecting between design options, would be developed and would include risk mitigation features.

At the time when the DEMO design is initiated, there should be a well-defined set of design requirements and a well-documented, low-risk design approach based on a complete set of validated and qualified components and processes to be integrated into an acceptable-risk demonstration power plant.

3. Addressing safety and environmental aspects. Of the many integrated design, modeling, and simulation aspects important for fusion power development, comprehensive safety and environmental evaluations in the context of advanced design studies are essential for proving the key benefits of fusion power relative to other energy systems. Fusion power is being developed under the very restrictive "no-evacuation criteria," very little decay heat, recyclable and clearable materials, and no long-lived waste products, to name a few.

Simulation tools are necessary to provide the protective assurances for the regulators, power producers and the general public. These tools must be developed, refined, and validated specifically for fusion developmental facilities and eventual power systems (as needed for licensing). The tools include robust computational models considering integrated thermal, chemical, and electromagnetic phenomena possible during postulated events, single-effects and integral experimentation to validate the physical models, a consistent method for interfacing safety analyses with the system design processes, and a suitable framework for safety assurance and regulation within parties considering fusion power deployment. For example, scientific understanding is presently incomplete and simulation tools are subsequently inadequate for evaluating the magnetic energy interaction, or arcing, during material failure. The behavior of tritium in irradiated structural, plasma facing, and functional materials is not well understood at present. This issue is closely related to similar issues examined in the ReNeW Theme "Taming the Plasma-material Interface." The safety relevance is found in the potential, and presently uncertain, estimates of tritium, dust, and other dispersible materials released in postulated accident scenarios. Simulation tools for safety analysis and materials performance must be integrated to address this issue.

There is a need to establish an integrated management strategy that could handle the sizable amount of mildly activated materials anticipated for fusion power plants. Because of concerns about the environment, limited capacity of existing repositories, high disposal cost, and political difficulty of constructing new repositories, the geological disposal option should be avoided and more attractive scenarios should be considered:

- Recycling and reuse of activated materials within the nuclear industry.
- Clearance or free release to the commercial market, if materials contain traces of radioactivity.

There is a growing international effort in support of this new trend and several fusion studies indicated recycling and clearance are technically feasible. The US fusion program should accommodate this new strategy. A dedicated R&D program could address the issues identified for each option. Fusion designers must minimize the radioactive waste volume by clever designs, and the materials community should continue developing low-activation materials and accurately measure and reduce impurities that deter the free release of in-vessel components.

Integrated Modeling

Improved integrated models that utilize advanced computational simulation techniques to treat geometric complexity, and integrate multi-scale and multi-physics effects, will be key tools for interpreting phenomena from multiple scientific disciplines and fusion experiments while providing a measure of standardization in simulation. The vision of such an effort is to provide higher predictive accuracy and substantially reduce the risks and costs in the development of fusion nuclear systems. These models would be able to predict the integrated behavior of tritium fuel cycle processes as well as of fusion power components in the fusion environment. As an example, the attractiveness of the dual-coolant lead-lithium breeding blanket depends on its ability to capture a large portion of the nuclear energy in the lead-lithium stream and transport it at high temperature with low pumping power to the power conversion system. The degree to which this is achievable depends on the fluid flow phenomena in the lead-lithium, which are highly coupled to the interactions with the magnetic fields, material properties of the flow channel insert, geometry of the design, deposition of the nuclear energy, etc. Integrated predictive capabilities can be instrumental in: evaluating design options, providing information for which diagnostic sensors are limited and experimental access is difficult; developing knowledge to focus and minimize required experiments; and interpreting the information gained.

Simulations will progressively allow for design optimization, performance evaluation, failure mitigation, and operational control of ITER and FNSF components in the near term and DEMO in the future. Initially, simulation development will focus on the component level modeling; subsequently, component level analyses will be integrated with system level modeling for global performance and safety analyses. The development of the integrated models will be pursued with the mindset of providing a link to the Fusion Simulation Project (FSP). Ultimately, the vision is development of a predictive capability for DEMO, after strong benchmarking by experimental data obtained from the "real fusion environment" on ITER and FNSF.

Integrated model development will be built upon four fundamental undertakings.

1. Integration and assimilation. The first activity will be the integration and assimilation of state-of-the-art analysis codes from the various fusion disciplines involved including neutronics, electromagnetism, plasma-material interaction, thermo-fluids, species transport, structural mechanics, and off-normal transient phenomena. There exist analyses codes that cater to these individual physics, but they have to be enhanced and tuned for application in the fusion plasma chamber environment. Examples include enhancement of liquid metal magnetohydrodynamic (MHD) codes and of plasma facing material behavior under transient energy depositions.

2. Mapping across various analysis codes. The second endeavor includes advances in data translation, involving efficient and high-fidelity data mapping across various analysis codes, enabling integrated or coupled simulations in a multi-physics environment. Numerical modeling of individual physics has its own unique mesh resolution requirements, and in most realistic calculations, the computational meshes used for different physical analyses differ in nature. The integrated suite must exchange data in a seamless and error-free manner and be compatible with modern clusters and parallel execution. The modeling of accident scenarios in the fusion reactor is a typical example requiring synchronized simulation of the effects of an individual component with the entire system.

3. Computational analysis management. The integrated modeling simulation process management system includes: storing relevant simulation data; transmitting them to multiple solvers in an appropriate format; and making the results available for post-processing, visualization and debugging utilities. A multi-physics integration arises from the ability to perform all analyses on geometric models derived from an identical representation, i.e., the Computer-aided Design (CAD)-based solid model. This common domain representation points to a strategy for expanded multi-physics applications where the internal representation of the geometry is common across the simulation tools.

4. Verification and validation. The integrated model development must reflect true plasma chamber system behavior. Model validation will arise from comparison with the existing experimental data, particularly data from experiments that address multiple effect and material interaction tests. During the ITER construction phase, the integrated model could be heavily utilized for mock-up and TBM designs. Ultimately, this integrated simulation tool, strongly benchmarked with the experimental data obtained from the "real fusion environment" on ITER and FNSF (Component Test Facility), will evolve to a validated predictive capability for DEMO.

Scale and Readiness

ARIES forms the basis and starting point for the integrated design activities for future magnetic fusion energy power plants, although no specific activity related to DEMO has been commissioned. More extensive design activities could be initiated at any time. The scope and level of effort should be expanded to approximately $10M per year as the DEMO design activities progress from conceptual design, to a more comprehensive integrated design effort including detailed RAMI, safety and environmental aspects. This level of effort is needed to more fully engage the broad spectrum of expertise required to carry out such detailed activities.

The effort to develop an integration model coupled with a computer-based geometric component has begun. Preliminary work involving coupling neutronics and thermo-fluids analysis codes based on a common CAD model, has been applied to the US ITER first wall/shielding blanket design. However, a focused effort of the development of an integrated model for fusion chamber specific research is yet to be initiated. Development of mature fusion specific physics models and code integration tasks would initially require funding at the level of $1-2M. Collaboration with the experts in the areas of computer science and applied mathematics from the FSP team would provide consistent, convergent, accurate solutions to the coupled multi-physical problems. As the project progresses, efforts should expand to produce experimental data for code validation.

Benefits of Integrated Designs and Models

Development of effective fusion power is a cost-intensive process, fraught with many compounded complexities and increasingly expensive experiments and test facilities. Any experiment or facility failure would be very costly to the fusion program. Thus, major failure risks must be aggressively mitigated. These integrated models and designs offer a cost-effective risk-mitigation option for the fusion program at a fraction of the cost of a single facility or experiment. It also enables a wide range of design exploration and optimization over a broad spectrum of options and configurations. Once an acceptable approach is selected, these tools will allow fine-tuning of the design and processes necessary to effectively design, build, and operate DEMO. Additional benefits are obtained from recycling techniques and guidelines for the free release of fusion-specific materials, both of which are beneficial to fission industry and US regulatory organizations.

Other Scientific Benefits

Integrated modeling continues to produce advances on research, knowledge, and technology for all science fields. Given the complexity of the fusion geometrical scale and multi-physical phenomena, development of a computer-based simulation capability to display fusion plasma chamber behavior poses many challenges and opportunities for innovations in mathematics, computer graphics, and architectures. These advancements will lead to further development of efficient multi-physics algorithms, methods, and techniques.

Connections to Other Thrusts

The integrated DEMO design activities, by definition, must strongly couple to all aspects of the fusion program and the thrusts that support them, since DEMO provides a focus in guiding fusion R&D. These include, for example, coupling to physics thrusts on alpha physics (3), plasma control (5), plasma dynamics (8) and transient event control (2), to the superconducting magnet thrust (7), to plasma edge and interface thrusts (9-12), and to fusion power and material thrusts (13, 14). The FNSF activities couple to Thrust 13 where a progressively more integrated modeling and experimental program on power extraction and closing the fuel cycle leads ultimately to fully integrated experimental validation in an FNSF. They also couple to Thrust 16 where a spherical tokamak configuration is proposed for FNSF.

The integrated model provides an effective mechanism in integrating models from other thrusts as well as ongoing R&D results in creating powerful and effective simulation tools. These specific models include those mentioned in Thrust 13 for power extraction, plasma fueling and tritium recovery as well as the detailed material modeling described in Thrust 14. The development of the integrated model would closely follow the development of the FSP, covered in Thrust 6, to ensure compatibility. For example, the model when linked to the simulation capabilities being envisioned in the FSP would be able to simulate the responses of the plasma facing surface, as well as the corresponding in-vessel component behavior, to various fusion plasma shots.

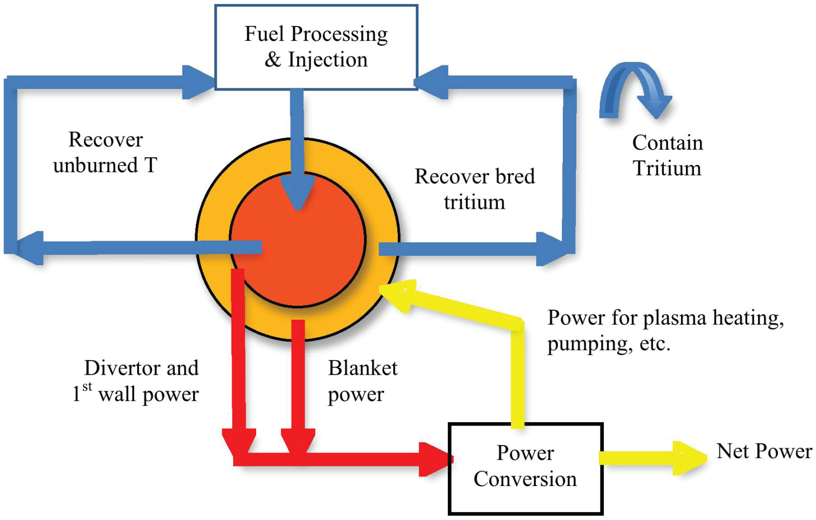

Figure 1. Schematic of key fuel cycle and power extraction systems.

Materials and Components:	**Fuel Cycle:**	**RAMI:**
Surface heating	Tritium breeding ratio	Radiation dose
Nuclear heating	Burn fraction	Component life
Temperatures	T tie-up in components	Failure modes, MTBF
Temperature gradients	Vacuum pumping	Maintenance processes
Mechanical stress	T control and recovery	Availability
Thermal stress	Processing	Diagnostics and inspection
Response to disruptions	Fuel injection	Mean time to repair/replace
Effects of neutron damage	T accountability	
Transmutation	T containment	
Gas production		
Power Extraction and Conversion:	**Safety and Environment:**	**Integrated System Model:**
Fusion power	Activation	Plasma physics
Surface & nuclear heating	Source terms	Physics and eng constraints
Duty cycle	Accident scenarios	Capital and operating costs
Coolant flow in B field	T inventory	Financial parameters
Conversion efficiency	Radwaste level	Plant layout
Plasma heating	Recycling and free release	Aspects from other lists
Recirculating power	Licensing	
Net power output		

Table 1. Major systems design aspects and processes that must be simulated.

Thrust 16: Develop the spherical torus to advance fusion nuclear science

The spherical torus (ST) is a low aspect ratio (low-A) tokamak that offers unique physical properties due to its very strong magnetic curvature and compact geometry. This configuration delivers high plasma pressure relative to the external magnetic pressure, and strongly affects plasma stability and confinement. It offers the promise of simple magnet design, reduced size, cost, and ease of maintainability. The ST program has made substantial progress. Non-solenoidal startup currents have reached 25% of required initiation levels. A strong favorable dependence of plasma thermal confinement on magnetic field was found. A five-fold increase in electron confinement was demonstrated in a small device with liquid metal walls. Stability needed for a component testing device was shown for several current relaxation times, with a majority of the current provided noninductively. The ST program is now poised to generate the knowledge base to confidently construct and operate a low-A Fusion Nuclear Science and Technology (FNST) component testing device, and to aggressively pursue improvements to advance the ST for energy production.

Key Issues:

- The ST has little room for a central solenoid to produce and drive plasma current. *Can plasma current be initiated and raised to high values without a solenoid?*
- The compact geometry of the ST increases the heat loading to the wall. *Can plasma exhaust power be effectively dissipated in the ST?*
- The ST energy confinement improves faster with magnetic field and plasma temperature than at high aspect ratio. *Can predictive models for ST confinement be developed and validated?*
- The broad current profiles and near-spherical geometry of the ST strongly affect stability. *Can stable and continuous operation be produced at low aspect ratio?*
- The lower magnetic field and enhanced energetic particle drive of the ST may challenge the sustainment of high plasma current. *What are viable means to maintain the current and control the plasma profiles in the ST?*
- The compact geometry of the ST precludes the use of shielded superconducting magnets. *Can suitable magnets be developed for ST applications?*

Proposed Actions:

- Exploit and understand magnetic turbulence, electromagnetic waves, and energetic particles for megampere plasma current formation and ramp-up.
- Develop innovative magnetic geometries and first-wall solutions such as liquid metals to accommodate multi-megawatt per square meter heat loads.
- Utilize upgraded facilities to increase plasma temperature and magnetic field to test the understanding of ST confinement and stability at fusion-relevant parameters.
- Implement and understand active and passive control techniques to enable long-pulse disruption-free operation in plasmas with very broad current profiles.

- Employ energetic particle beams, plasma waves, particle control, and core fueling techniques to maintain the current and control the plasma profiles.
- Develop normally conducting radiation-tolerant magnets for low-A applications.
- Extend the ST to near-burning plasma conditions in a new or further upgraded device.

A strongly integrated plasma theory and modeling effort, validated against experiment, is required to enable an optimal ST design for a component test facility. This Thrust will also benefit from, and contribute to, the understanding of the conventional aspect ratio tokamak.

The spherical torus is a low aspect ratio tokamak offering unique physical properties due to very strong magnetic curvature and compact geometry. It has potential advantages in size, cost, maintenance, and magnet simplicity. It produces high plasma pressure relative to the external confining magnetic pressure, and promises high neutron wall loading with reduced tritium consumption. The ST program has made substantial progress since its inception, and is now ready to develop the knowledge base to confidently construct and operate an ST-based FNST component testing device, and to aggressively pursue improvements to advance the ST for energy production. This knowledge base is embodied in the Thrust elements below that describe: (1) plasma startup and ramp-up, (2) plasma-material interface (PMI), (3) transport, (4) plasma stability and control, (5) current sustainment, (6) magnets, and (7) next-step ST facilities.

Thrust 16 Elements:

1. Exploit and understand magnetic turbulence, electromagnetic waves, and energetic particles for megampere plasma current formation and ramp-up.

Low-A tokamak designs reduce or eliminate the available transformer flux for plasma current initiation and ramp-up to reduce device size and aspect ratio. Near-term startup research should continue to focus on developing the understanding of magnetic helicity injection, radiofrequency wave-based startup, and novel inductive startup schemes. Noninductive ramp-up to full operating current requires additional effort beyond the capabilities of present STs. Physics-based modeling of plasma current ramp-up must be validated against existing and upgraded ST experiments, and be extended to include the effects of energetic particle instabilities on fast-particle transport. These validated models are needed to specify the required noninductive current drive capability of future ST experiments.

Actions:

- Develop helicity-injection-based startup to a current level sufficient to provide a target for radiofrequency or neutral beam (NB) ramp-up in present and upgraded facilities.
- Perform, in present or upgraded facilities, startup and initial buildup of a discharge using radiofrequency in the electron cyclotron frequency range to a similar level. Cost-effective collaborative experiments on the DIII-D tokamak device should be explored.
- If needed, conduct mechanical and thermal testing of neutron-tolerant inductive systems for startup assist, including small iron core and mineral-insulated solenoid approaches.

- Develop physics-based, self-consistent, integrated models of startup, with sufficient predictive capability to permit an evaluation of the most promising startup approaches.

- Develop a similar level of predictive capability for ramp-up systems, including radiofrequency and neutral-beam-based systems.

- Implement noninductive (or small-induction, if appropriate) startup, at a toroidal field, B_T, of order 2 tesla, and ramp-up to the multi-megampere current level.

Links to other thrusts: Theme 2, high-performance steady state; auxiliary systems (Thrust 5)

2. Develop innovative magnetic geometries and first-wall solutions such as liquid metals to accommodate multi-megawatt per square meter heat loads.

a. Develop and understand innovative magnetic geometries and particle control.

While many of the plasma-material interface issues are common to all high-temperature magnetically confined plasma devices, the compact geometry of the ST and operation at low normalized density define a unique edge transport regime with much greater demands on divertor and first-wall particle and heat flux handling. Normal and transient heat and particle flux mitigation, and control strategies beyond those used in present devices, and/or envisioned for near-future devices, such as ITER, must be developed. These solutions must integrate favorable edge pressure levels (pedestals) with simplified future remote handling capabilities.

Actions:

- Develop divertor and first-wall solutions to reduce steady-state peak heat fluxes from the projected level of 20-60 MW/m^2 to ≥ 10 MW/m^2 and transient loads to ≥ 10 MJ/s extrapolable to a high-duty cycle nuclear environment. Potential solutions include new magnetic divertor configurations such as the X, Super-X, and “Snowflake” divertor, volumetric momentum and power exhaust, and innovative plasma facing components, e.g., liquid metal or “pebble” divertors. These configurations should be tested in a low-collisionality edge (scrape-off) layer relevant to ST applications in existing and upgraded facilities. However, new ST devices may be required for demonstration of integrated long-pulse performance.

- Develop efficient impurity, helium and hydrogenic density control at levels 20-30% of the empirical (Greenwald) density limit, in combination with the proposed heat flux mitigation techniques. Presently envisioned pumping solutions include cryopumping or low recycling liquid metal wall and divertor.

- Develop efficient plasma refueling techniques, including neutral beam fueling, pellet and compact toroid injection, and high-density gas jets, for continuous low normalized density, high confinement mode (H-mode) operation with an appropriate edge plasma pressure.

- Develop and validate theoretical and numerical models of edge transport and turbulence in the ST, including new codes with gyrofluid and gyrokinetic models. To help validate the models, develop and implement new diagnostic measurements to elucidate the edge electron and ion distribution functions, edge flows, and edge neutral density.

Links to other thrusts: Theme 3, plasma-material interface (Thrusts 9-12)

b. Develop and understand liquid metal plasma facing components.

Liquid metal plasma facing components (PFCs) may advance the ST by: 1) significantly reducing anomalous electron transport, 2) allowing operation at high wall fusion power loading, 3) providing density control, 4) permitting density and pressure profile control at high beta (via the fueling profile), 5) providing edge and core stability enhancements, 6) controlling high-Z impurities (if utilizing liquid lithium), and other issues common to conventional tokamaks. High recycling liquid metal PFCs such as gallium may be better suited than lithium to high-power density divertor designs. However, the technology requirements for liquid metal PFC development are largely independent of the choice of liquid metal. The near-term deployment of liquid lithium PFCs will inform the development of high recycling liquid metal PFCs as well.

Actions:

- Implement and evaluate full liquid lithium wall in an ST, with full core neutral beam fueling to assess if high ohmic energy confinement can be extended to core-fueled, auxillary-heated STs.

- Implement and investigate a full liquid lithium divertor in an ST with full core fueling to assess whether a full lithium wall is required, or if partial walls or a divertor alone are sufficient.

- Diagnose edge plasma effects and plasma-material interactions with liquid lithium walls. Use this data to validate models for the plasma edge, and core-edge coupling, over a range of global recycling coefficients. Understand the effect of low recycling on the edge and core plasma sufficiently well to project the results to larger, hotter devices.

- Implement and evaluate a full flowing liquid metal wall and divertor in an ST, with pulse length comparable to the flow time for liquid metal from inlet to outlet, over the liquid metal “former” or guide wall, jet path, etc. Evaluate plasma magnetohydrodynamic (MHD) effects on liquid metal flows, stability, and influx to the core plasma, to determine whether liquid metal walls can be implemented, and whether liquid metals can control recycling, in a fusion nuclear device.

Links to other thrusts: Theme 3, plasma-material interface — liquid metal test stand work, MHD simulations (Thrusts 10, 11)

3. Utilize upgraded facilities to increase plasma temperature and magnetic field to understand ST confinement and stability at fusion-relevant parameters.

Recent NSTX and MAST results indicate that the parametric dependencies of energy confinement in the ST differ from conventional tokamak scaling, creating the possibility of better than expected confinement at low-A. In particular, the energy confinement improves faster with increasing magnetic field and decreasing plasma collisionality in the ST than at high-A. The differences appear related to strong flow-shear suppression of turbulent ion transport in the ST to near neoclassical (collisional) levels, together with increased electron thermal transport. Since neoclassical ion transport and some of the instabilities thought to drive rapid electron transport, such as micro-tearing modes, have a strong collisionality and field dependence, it is important to understand how this transport scales to higher field and lower collisionality. The latter strongly affects the large trapped particle population in STs and is the dimensionless variable of largest extrapolation from present to future STs. Furthermore, with its relatively low viscosity and high flow rates, the low ion transport regime accessible in the ST offers a unique test of our understanding of ion transport in toroidal plasmas. In addition, a large gap exists between experimental observations and our understanding of energetic particle (EP) effects on plasma heating, current drive and transport. Importantly, the energetic particle populations of present STs have ratios of fast ion velocity to Alfvén velocity similar to those of the alpha populations in tokamak reactors. Thus, ST EP research is highly relevant for burning plasmas and ITER.

Actions in the area of thermal transport:

- Determine confinement trends in the ST over an extended range of collisionality, current, field, power input, aspect ratio, and wall recycling conditions. Studies would begin in planned upgraded STs (near-term) and could require new ST experiments with engineering parameters several times those of present devices for confident extrapolation to next-step devices.

- Understand the causes for anomalous electron transport utilizing advanced diagnostics for high and low wavelength, electrostatic and magnetic turbulence in the planned upgraded STs, including diagnosis of electron temperature gradient streamers and micro-tearing instabilities.

- Measure the ion and electron velocity distribution function in the ST to supplement power-balance transport assessments.

- Develop a predictive understanding of anomalous transport in the ST utilizing numerical codes (including synthetic diagnostics) and transport control tools to optimize the performance of future reactors — in particular, optimizing the aspect ratio.

Actions in the area of energetic particle transport and their interaction with the background plasma:

- Develop numerical and theoretical models for energetic particle transport in the presence of multiple instabilities arising from the large normalized gyroradius, ρ^*, of fast ions in the ST. This key parameter will differ from present-day STs by a factor of 2 in a component test facility (CTF) and 4 in burning plasma STs, challenging predictive models. Experiments at intermediate ρ^* (e.g., CTF) are required to extrapolate to burning plasma DEMO-class STs.

- Develop new diagnostics for internal oscillations and EP profile measurements in ST plasmas to establish the database of EP loss and transport, for both present-day and next-step STs.

- Engage in theoretical investigation of the interaction between multiple Alfvén modes and the background plasma. Measure the Alfvén mode structure and the effects of these modes on thermal ion heating and electron transport. Investigate their control using external antennas as a possible means to control the channeling of the EP energy into thermal electron/ion energy.

Links to other thrusts: Theme 1, burning plasmas and ITER: alpha particles; Theme 2, high-performance steady state: validated modeling (Thrusts 3, 6)

4. Implement and understand active and passive stability control techniques to enable long-pulse disruption-free operation in plasmas with very broad current profiles.

Maintaining the continuous high-pressure plasma operation required for efficient ST-CTF and DEMO operation at a low level of neutron fluctuation and plasma disruptivity requires an understanding of stability and control at reduced levels of collisionality, and broader current profiles, than attained in present devices. A spherical torus component test facility will not perform this research, so understanding must be developed, and performance levels reached now, for both CTF and DEMO-class applications. Compared to tokamaks, the ST operates in a unique parameter regime of very low internal inductance, high pressure normalized to magnetic field (beta), and high ratio of fast ion to Alfvén velocity. Physical effects in this regime, including distinct global instability structures, increased multi-mode effects, explicit geometric dependence of neoclassical tearing mode (NTM) stability, and magnetic control of plasma rotation, require dedicated ST research.

Actions:

- Reduce the plasma internal inductance via broader neutral beam injection/radiofrequency current deposition and/or lithium walls, aiming for self-consistent current profiles to test kink/ballooning and resistive wall mode (RWM) stability near the current-driven kink limit. Assess elevated safety factor $q > 2$ for NTM stability and provide a wide range of accessible beta since the current-driven kink may be unstable at any normalized beta, and multiple modes may be important at high normalized beta.

- Reduce plasma collisionality up to an order of magnitude utilizing ST device upgrades to test theoretical RWM stabilization and the expected increase of nonresonant magnetic braking.

- Develop control of the plasma rotation and shear using expanded NBI and nonresonant magnetic braking capabilities to understand the effect of rotation and shear on RWM and NTM.

- Implement expanded active RWM feedback control (e.g., non-magnetic sensors, multi-mode capability, upgraded 3-D control fields), dynamic error field correction, moderate

3-D shaping, and beta control to significantly reduce disruption probability and variations in fusion reactivity.

- Demonstrate sustained operation at normalized beta values beyond ST-CTF, approaching DEMO levels, to reduce performance risk, and rapidly achieve neutron fluence goals.
- Develop validated computational tools for the experimental research actions.

Links to other thrusts: Thrust 2 (transient/disruption control), Thrust 5 (continuous, high-performance operations), Thrust 6 (predictive modeling), Thrust 17 (3-D fields for stability), Thrust 18 (low field)

5. Employ energetic particle beams, plasma waves, particle control, and core fueling techniques to maintain the current, and control the plasma profiles.

The sustainment of high-performance integrated ST scenarios requires full noninductive current drive and sufficient control of the plasma profiles. Future ST applications are expected to rely on a combination of current drive sources, including the neoclassical bootstrap current, NBI current drive, and plasma wave current drive (fast-wave, electron Bernstein, and/or electron cyclotron). Spherical torus operation at high plasma current (4-10 MA) will require access to a new ST parameter regime of sustained low collisionality, high thermal confinement, and high beta to provide high current drive efficiency and high bootstrap fraction. This regime must be self-consistently accessible with the heating and current-drive techniques proposed for ST applications — including during plasma current startup and ramp-up. Control of the NBI and wave current drive can in principle be achieved by varying injection parameters to vary the power deposition profile. However, the bootstrap current profile is largely determined by the plasma thermal and particle transport. Thus, new control techniques must be developed to optimize and ultimately control the plasma pressure profile to control the current profile.

Actions:

- Access one to two orders of magnitude lower collisionality by increasing plasma temperature with a two to four-fold increase in magnetic field, plasma current, and heating power.
- Implement particle pumping to achieve factor of two to four reduction in normalized density to support access to very low collisionality values. Pumping techniques for access to very low recycling regimes should also be investigated.
- Assess and optimize the impact of reduced collisionality on core and edge transport — in particular, the impact of transport on the bootstrap current density profiles.
- Test the ability of improved pumping and low recycling, combined with the development of deep core fueling, to enable modification and control of the core transport and pressure profile (and therefore the bootstrap current profile) for plasma sustainment and optimal stability.

- Increase sustained noninductive current drive capability to 100% with 50-70% bootstrap fraction. The current drive systems and profile control techniques should enable tests of sustained q above 2 and control the magnetic shear to optimize the stability and confinement.

- Increase plasma pulse length by at least one to three orders of magnitude to demonstrate sustained control of fully noninductive ST plasmas for many current relaxation times.

Links to other thrusts: Thrusts 2, 3, 5, 6, 9, 10, 11, 12, 15

6. Develop normally conducting radiation-tolerant magnets for low-A applications.

The inability to shield the central toroidal field (TF) coil in an ST from neutron bombardment motivates the development of neutron tolerant, insulator-free, low impedance, TF centerpost magnets. The simplest approach is to employ a single-turn, resistive, intensely cooled copper centerpost, driven by a low impedance current source such as a homopolar generator. Tight coupling of the generator output to the TF is required to minimize resistive losses. Presently envisioned applications for the ST emphasize neutron production, and since fusion power scales as B_T^4, systems with higher toroidal fields than available in present devices should be developed.

Actions:

- Conduct engineering studies for single-turn centerpost TF magnets. Evaluate insulator-free, multiple-turn TF systems, using air-gapped conductors, resistive shims, or other approaches.

- Conduct engineering studies of integrated, low impedance current sources for the TF system.

- Design and construct a candidate test TF magnet set. Perform electrical, mechanical, and thermal testing of the system, including generator(s), bus work, joints, and the coil proper.

- Conduct design and engineering studies for radiation-tolerant compact ohmic heating (OH) systems to determine if inductive startup is practical for a nuclear ST.

- Design and construct candidate OH system, and test electrical, mechanical, and thermal properties.

- Test radiation tolerance, displacement per atom limits, tritium migration into coolant channels, etc., for TF and OH systems. Employ fission neutron sources, International Fusion Materials Irradiation Facility (as available), and tritium facilities.

Links to other thrusts: Thrust 16 (element 1), Thrust 7.

7. Extend ST performance to near-burning-plasma conditions.

Present and upgraded ST facilities can go far in developing the knowledge base needed for ST fusion nuclear science applications. A factor of five to ten reduction in collisionality and increase in pulse duration should be achievable by doubling of the field, current, and heating and current drive power in upgraded ST facilities with a modest increase in aspect ratio ($A \geq 1.3 \rightarrow 1.5$). Depending on results from upgraded ST facilities and the worldwide tokamak research program, a new ST device with increased performance and capabilities may be needed to support the design and operation of a low-A device suitable for fusion nuclear science applications.

Actions:

- Further reduce collisionality to near-burning-plasma conditions to assess: current drive for ramp-up and sustainment at high current, core and pedestal transport, MHD stability, PMI solutions.
- Operate a high-performance ST for very long pulse lengths with actuators relevant to a high neutron fluence environment to assess: (1) sustained plasma current drive and profile control, reliable disruption prediction, avoidance, and mitigation for integrated ST conditions; and (2) compatibility of sustained high performance with high power and particle exhaust mitigation techniques, with equilibrated divertor and first-wall conditions and low hydrogenic retention.
- Test high-field, long pulse magnets under conditions directly relevant to ST applications.

Summary of Thrust 16

The Thrust elements provide a comprehensive set of research actions to advance the ST configuration to be ready to contribute to fusion nuclear science applications. These actions also enable access to a unique plasma parameter regime of high normalized plasma pressure, low collisionality, and low aspect ratio. This Thrust enhances the understanding of compact alternative magnetic configurations, expands the understanding of conventional aspect ratio tokamaks including ITER, and develops capability necessary to prepare for a fusion DEMO.

Thrust 17: Optimize steady-state, disruption-free toroidal confinement using 3-D magnetic shaping, and emphasizing quasi-symmetry principles

The stellarator concept relies on currents in external coils to confine plasmas magnetically. Stellarators can therefore operate continuously if supplied with heating power and effective means of heat and particle exhaust (divertor). They have experimentally demonstrated sustained plasmas with good confinement, high normalized pressure (β), and do not suffer from virulent current or pressure-driven instabilities that abruptly terminate the plasma.

The magnetic field in a stellarator is not symmetric in the toroidal direction, but has three-dimensional (3-D) structure. Three-dimensional shaping provides additional control of plasma confinement not available in the axisymmetric tokamak, and permits designs that are passively stable to major instabilities, with minimal feedback control. The lack of symmetry in a conventional stellarator, however, can reduce the confinement of energetic ions, including alpha particles. The application of innovative **quasi-symmetric (QS)** shaping is predicted to resolve this issue, leading to stellarator designs that confine high-energy particles, and permit plasma flows (also favorable for plasma confinement), while retaining the robust stability of the stellarator. The QS stellarator is thus a transformational concept, offering a timely, effective solution to the challenges of severe transient events and control in steady-state, high-pressure plasmas. In addition, 3-D fields affect all configurations through self-organization or external perturbations. Understanding of 3-D effects is thus a core competence, required for the success of all magnetic configurations.

There is a need to understand 3-D shaping in an integrated manner in plasmas with higher levels of performance.

Key Issues:

- Simultaneous achievement of QS confinement at high-pressure without disruptions. *Does quasi-symmetry lead to improved confinement at high plasma pressure? Is there an optimal type of QS shaping?*
- Three-dimensional shaping requires magnets that are more complex than planar coils. *How can optimized magnet design ease construction, reliability, cost and maintenance of 3-D systems while meeting the physics requirements?*
- Divertors for 3-D configurations require development. *How do we integrate an effective divertor into the optimization of quasi-symmetric stellarators?*
- Three-dimensional fields have application to all toroidal systems. *What is the optimum amount and type of 3-D shaping to effect improvements?*

Proposed Actions:

1. Conduct two quasi-symmetric experiments spanning a broad range of internal plasma current, with plasma parameters sufficient to demonstrate low collisionality, disruption-free operation at high plasma pressure. Examine the merits of completing NCSX as part of this effort. Expand efforts in non-axisymmetric theory and computation to develop predictive models of QS confinement. Extend the understanding of QS plasmas to near-burning conditions.

2. Investigate quasi-symmetric configurations with simpler and maintainable magnet systems.

3. Design 3-D divertors compatible with QS geometry. Integrate with 3-D coil simplification.

4. Explore the addition of 3-D shaping to other magnetic configurations.

Scientific and Technical Research

A fusion power system that operates continuously without frequent destructive off-normal events is a desirable goal. Stellarator devices already demonstrate sustained plasma performance without abrupt terminations or transients[1]. They also achieve good confinement of high-density plasmas, typically at temperatures lower than in tokamaks, but not far below that required for fusion applications.

The geometry of the stellarator is an example of 3-D magnetic shaping. The magnetic fields confining the plasma do not possess symmetry in the toroidal direction, as nominally exhibited in tokamaks and most other fusion plasma devices. The 3-D approach produces the necessary helical magnetic field with magnet coils located outside the plasma. An internal toroidal plasma current is not required, thus avoiding the need for auxiliary current-drive systems and rapid control of the current and pressure profiles to maintain stability. In addition to providing rotational transform (parameter measuring the helicity of the confining field), 3-D shaping also provides control over global and local shear and curvature of the field, the magnetic well depth, the location and fraction of trapped particles, shear of the radial electric field, and plasma viscosity. In short, 3-D shaping may provide control of not only neoclassical transport and energetic particle confinement, but also the geometrical parameters that affect turbulent transport and global stability. However, the conventional stellarator's lack of symmetry can lead to inadequate confinement of hot fusion plasmas, and of the energetic charged fusion products in particular. Remarkably, 3-D magnetic configurations can be designed to be quasi-symmetric to reduce such losses. Even though the magnetic field of the QS stellarator plasma remains three-dimensional, the confinement of plasma particles within it may be similar to that of an axisymmetric device such as a tokamak. A QS stellarator fusion system would thus operate steady state, without threat of disruption, with comparatively few control needs — and with adequate confinement of energetic fusion products.

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[1] *Presently, stellarators do not exhibit disruptions in normal operation. In contrast to tokamaks, pressure limiting behavior is not found to terminate the discharge. Nonetheless, stellarator plasmas can be abnormally terminated by radiative decay of the plasma energy triggered by excessive density or material falling into the discharge from the wall. In both of these cases, the discharge decays on the relatively long scale of an energy confinement time.*

The improved confinement from quasi-symmetry has been demonstrated in a university-scale device, and 3-D theory and modeling continues to develop the range of QS confinement possibilities. This Thrust calls for new experiments combined with advances in 3-D theory, to build QS stellarators that operate at higher temperature and plasma pressure. The goal of this effort is to develop sufficient scientific understanding to assess the feasibility of a burning plasma device based on the QS stellarator. If successful, the approach could obviate the need for most of the plasma control sensors and actuators used or proposed for tokamak operation, and largely avoid the risks associated with operating fusion plasmas at or beyond their stability boundaries.

Research Elements

Progress on stellarator experiments in Japan and Germany over the past two decades leaves little doubt that high-performance, sustained, disruption-free plasmas are attainable in 3-D configurations. This Thrust delineates activities to advance our understanding of QS stellarators to realize this level of performance. Elements of the Thrust also address the unique challenges of constructing the magnets of 3-D devices, including the implementation of 3-D divertors. Lastly, even tokamaks and other nominally axisymmetric toroidal systems increasingly exhibit nonsymmetric equilibria, and could benefit from varying degrees of 3-D shaping guided by analytic and numerical tools drawn from stellarator research.

The Thrust actions are organized into four elements:

- New QS stellarator experiments for improved confinement in high-performance plasmas.
- Design and construction of 3-D coil systems.
- Divertors for 3-D configurations.
- Three-dimensional shaping for improved operation of other toroidal systems.

Action 1: New QS stellarators for improved confinement in high-performance plasmas

Promising results obtained from the exploratory quasi-symmetric HSX device, coupled with the scientific benefits of quasi-symmetry, motivate pursuing QS confinement at a larger scale. The extensive knowledge base from larger (non-QS) stellarators gives confidence that a research program in quasi-symmetric 3-D confinement could be successfully addressed on an experiment at the performance extension (PE) scale — an integrated investigation of sustained, stable plasmas with hot ions, high pressure, and good confinement — as a precursor to a QS burning plasma device that could follow ITER. This Thrust as a whole outlines the key research steps to be taken in preparing for such a performance extension device. In addition to expanded efforts in theory and international collaborations, these include two intermediate-scale proof-of-principle experiments that will provide the span of scientific and technical knowledge required for the penultimate step to a burning QS stellarator plasma.

The key areas in which new knowledge is required are:

- Integration of high β (normalized plasma pressure) performance with good confinement and without transient events; understanding the physics underlying the benign nature of pressure limits in stellarators, and the impact of pressure-driven currents on maintaining quasi-symmetry.
- Demonstration of the fundamental benefit predicted of QS: tokamak-like confinement of energetic ions in high-temperature, collisionless plasmas.
- Understanding of reduced viscous damping of plasma flows in QS plasmas and its relation to higher plasma confinement through suppression of plasma turbulence, similar to tokamaks.
- Control of impurity accumulation and helium ash expulsion. Despite predictions that stellarator plasmas exhibit impurity accumulation, experiments have realized high-confinement regimes in which impurity confinement is acceptably low. Extension of these results to higher temperature QS plasmas is required.
- Three-dimensional divertors for QS configurations. Validation of 3-D physics models of divertor performance and development of appropriate coil configurations.

Different types of quasi-symmetric 3-D shaping offer different trade-offs. Quasi-axial (QA) symmetry is similar to the axisymmetry of the tokamak, and, with finite plasma pressure, gives rise to moderate levels of self-driven current parallel to the magnetic field (bootstrap current). Quasi-axial symmetry would provide a direct comparison of quasi-symmetry to true symmetry, and may answer why the density and pressure-limiting behavior is substantially different in stellarators and tokamaks. Quasi-helical (QH) and quasi-poloidal (QP) symmetry exhibit lower levels of bootstrap current than the tokamak-like QA configuration, rendering them less susceptible to current-driven instabilities and the need for external control.

Coordinated pursuit of both the low (QH, QP) and moderate current QA approaches would establish the efficacy of QS compared to true symmetry, and determine the magnitude of stellarator 3-D shaping required to provide stable sustainment with finite plasma current. Results from these experiments combined with theory would inform the choice of the optimal type of quasi-symmetry for a **single** performance extension device.

Key research steps to provide the requisite knowledge include:

- Construction and operation of two intermediate-scale experiments: a QA device with moderate bootstrap current and a QH or QP experiment with low bootstrap current. Both experiments would have sufficient pulse length and heating power to evaluate β-limits at low collisionality to compare with theory. The two devices differ significantly in the details of their magnetic configuration, such as field line curvature, shear, and trapped particle fractions, allowing comparative tests of turbulent transport, impurity transport, and density limits. The two experiments could also differ in the magnet coil and 3-D divertor design.

- Experiments varying the magnitude of plasma current in plasmas of similar parameters that would address relative susceptibility to macroscopic instabilities and requirements for control. The QA experiment would extrapolate the plasma characteristics of symmetric tokamaks to 3-D systems.

- Expansion of 3-D theory and modeling in the areas of turbulent transport, β-limits, impurity transport, nonlinear effects, effects of stochastic magnetic fields, energetic particle effects, effects of plasma rotation, and kinetic effects on equilibrium and stability with 3-D fields.

- Inclusion of 3-D theory and modeling in the effort to develop predictive simulation capability for fusion plasmas (Thrust 6) applicable to both stellarators and tokamaks, in which small asymmetries of order $dB/B \sim 10^{-3}$ are known to affect the plasma behavior.

- Targeted collaboration on the PE-scale, nonsymmetric experiments LHD (Japan) and W7-X (Germany). These activities will focus on steady-state 3-D divertor performance, pressure-limiting mechanisms in stellarators (experimentally benign, but not yet well understood), and integrated performance, e.g., confinement of high-temperature, high-pressure plasmas with low impurity content.

- Implementation of research program to extend the knowledge of QS plasma confinement to near-burning conditions in a long-pulse, high-field QS PE-scale experiment. The outcome will be a predictive understanding of the dependence of QS confinement on system size and plasma temperature that extrapolates to burning plasma performance. The results of the other actions of this Thrust and further reactor studies will guide the design of this experiment.

Action 2: Design and construction of 3-D coil systems

The 3-D coil sets required to produce stellarator magnetic configuration are more complex than those used in tokamaks. The goal is to reduce the technical risk and cost of constructing and maintaining large-scale stellarators.

Key research steps include:

- Reevaluation of the plasma parameters (maximum β, degree of quasi-symmetry, aspect ratio, etc.) that drive the design of magnet coil. Evolving knowledge from experiments, theory, and engineering optimization will be folded into QS magnet design.

- Investigation through modeling of different coil geometries, e.g., continuous helical coils, saddle coils, range of aspect ratios, to identify desirable QS configurations with simpler coils.

- Greater use of auxiliary trim coils to ease fabrication and assembly tolerances, and increase flexibility in the magnetic configuration.

- Innovative use of magnetic materials to simplify the shaping of the 3-D field.

- Exploration of high-temperature superconductors, leading to steady-state magnets with relatively low operating costs, improved maintainability (demountability), and easier fabrication compared with conventional superconductors. Inherently steady-state devices such as stellarators would benefit greatly from this emerging technology (see Thrust 7).

The goal of this set of modeling and development activities is the development of a practical magnet system for a PE-class QS stellarator experiment with a predictable cost and schedule.

Action 3: Divertors for 3-D configurations

Magnetic field lines at the relatively cold edge of the plasma must be diverted to a region where helium ash from the fusion reaction and other impurities can be removed. The plasma temperature must be low enough to prevent rapid erosion of the plasma facing material in this divertor region. The high-density capability of even moderate-field stellarators makes a radiatively cooled divertor solution plausible, though the 3-D geometry makes the engineering design difficult. However, the understanding of divertor behavior in stellarators is less well-developed, and the adaptation of an effective divertor to 3-D geometry is more complex than in tokamaks. The island divertor concept employed on the W7-AS and LHD stellarators, and planned for W7-X, requires control of the edge rotational transform. It also constrains the divertor to adjoin the main confinement region. Three-dimensional divertor designs that require less edge plasma control and allow for expanded exhaust with rapid pumping are highly desirable. Such designs must also be integrated with the optimization of the entire stellarator magnet system.

Key research steps include:

- Designing advanced divertors that handle the necessary power and particle exhaust, and control neutral and impurity influx, while remaining compatible with QS 3-D shaping.
- Increasing participation on the large LHD and W7-X experiments in Japan and Germany in the area of 3-D divertor physics. This activity includes the benchmarking of 3-D edge physics transport modeling codes.

Action 4: 3-D shaping for improved operation of other toroidal systems

There is no clear demarcation between tokamaks and QA stellarators: the 3-D plasma shape can be continuously varied while maintaining quasi-symmetry. This action pursues the potential benefits of controllable **levels** of QA 3-D shaping to be pursued on tokamaks and other toroidal confinement systems:

- Providing sufficient poloidal magnetic field for sustainment of the magnetic configuration.
- Ensuring robust stability to prevent uncontrolled vertical displacements and disruptions.
- Minimizing the need for feedback systems.
- Understanding the empirical density limits of tokamaks by extrapolating from low-current 3-D stellarators (in which they are understood) to higher-current axisymmetric systems.

- Applying 3-D shaping to understand nonsymmetric quasi-single helicity (QSH) equilibria in reverse field pinches (RFPs).

Suppression of Type 1 edge localized modes (ELMs) in tokamaks (see Thrust 2) is required for ITER. While not explicitly a type of 3-D shaping, the successful application of 3-D perturbation fields to control ELMs makes use of analytic tools developed and used in stellarator research.

Key research steps include:

- Conceptual design and modeling of quasi-axisymmetric stellarators with variable levels of 3-D shaping to perform as stellarator-tokamak hybrids.
- Test of variable 3-D shaping on stellarator-tokamak hybrids.
- Application of 3-D analysis techniques to reverse field pinch plasmas.
- Application of 3-D analysis and design approaches to ELM suppression on existing tokamaks and ITER.

Understanding the fundamental consequences of 3-D shaping could bring wide-ranging benefits to the science of toroidal confinement.

Readiness

The US has the leading theoretical and experimental program in quasi-symmetric stellarators, and operates the only QS stellarator (HSX, with QH symmetry) in the world. A small stellarator-tokamak hybrid (CTH) investigates the suppression of current-driven instabilities with 3-D shaping. US researchers also are leaders in the control of ELMs by application of 3-D fields. In concert with a broad international program in stellarators, the US research program is scientifically and technically ready to proceed with diverted, hot-ion QS experiments at the proof-of-principle scale. The Thrust activities draw from:

- Positive results obtained on QS confinement from ongoing HSX experiments.
- Optimization procedures developed for the design of NCSX, a proof-of-principle scale QA experiment, providing a comprehensive suite of codes for the design of any new QS facility.
- The NCSX facility, which if completed would fulfill many of the scientific needs of the intermediate-scale QA facility described in this Thrust. Construction of this facility has been suspended, and serious consideration should be given to its completion and operation, subject to appropriate review.
- Knowledge of confinement scaling, stability, divertor performance, long-pulse sustainment, and limiting behavior in stellarators, obtained primarily from LHD and W7-AS.
- Existing and developing 3-D equilibrium and stability analysis tools.
- Ongoing experimental work on stellarator-tokamak hybrids.

- Availability of high-power radiofrequency (electron cyclotron heating) and neutral beam heating systems with proven application to stellarators

Scale and Integration of Efforts

The US stellarator program is prepared to make significant advances in the area of sustained disruption-free confinement. However, its experimental facilities are inadequate for testing the innovative ideas in QS confinement at high plasma pressure and temperature simultaneously. While the large international experiments can provide important information in this regard, they are not quasi-symmetric.

Two QS experiments with sufficient heating power to test ion confinement, plasma stability, limiting mechanisms, divertor performance, and integration of good confinement with high β would span the necessary range of operating space in quasi-symmetry and plasma current to extrapolate with confidence to an optimal PE-scale stellarator, and ultimately a reactor. They would be designed, built, and operated as a unified effort. This would ensure that the necessary scientific comparisons can be made, and that an appropriate range of coil design and divertor options, as well as heating systems, will be considered. The elements of this Thrust are clearly integrated toward this goal. In addition to construction costs, the effort will require manpower to be allocated to:

- Physics modeling and magnet coil optimization.
- Three-dimensional divertor design.
- Machine design and engineering.
- Physics planning.
- Diagnostics development and 3-D equilibrium reconstruction techniques.
- Integrated predictive theory.

Other Scientific Benefits

Now that 3-D effects are known to affect many toroidal confinement schemes, improved understanding of 3-D shaping is a broad scientific benefit to magnetic confinement. The Thrust to improve the integrated performance of a system that operates steady state without disruption addresses the clear need of fusion-scale devices to avoid severe off-normal events, as called for in the Greenwald Panel report. Techniques and analysis of nonsymmetric trim magnet coils are applicable to resonant magnetic perturbation coils for tokamaks in which urgent efforts are underway to understand and develop effective ELM-suppression approaches for JET and ITER. Lastly, the obvious application of high-temperature superconductors to steady-state stellarators could motivate more rapid development of that technology for all fusion systems requiring continuous high fields.

Connections to Other Thrusts

- Three-dimensional shaping of plasmas for steady-state disruptionless performance is central to the Theme 5 goal of optimizing the magnetic configuration. Three-dimensional shaping tools and analysis are applicable to other toroidal configurations described in Thrusts 16 and 18, as well as to tokamaks.

- With its emphasis on understanding sustained plasma configurations without virulent off-normal events, the Thrust contributes directly to the Theme 2 goal of creating predictable high-performance steady-state plasmas, providing a wholesale alternative to developing the integrated control strategies, sensors, and actuators discussed in Thrusts 1, 2, 5, and 8.

- Predictive capability is required for all fusion concepts. The need for 3-D predictive modeling embedded throughout this Thrust is reflected in Thrust 6.

- The potential benefit of HTS to use the inherent steady-state capability of stellarators motivates a linkage with Thrust 7.

- The effort to develop robust 3-D divertors, coupled with the need to understand power and particle handling, drives linkages with Thrusts 9-12.

Conclusion

The application of the novel principle of quasi-symmetry using 3-D magnetic fields seeks to combine the good confinement of the tokamak with the sustainment and robust macrostability of the stellarator. At present, 3-D shaping is the only known means of achieving these conditions, and could do so with minimal need for control sensors and fast actuators. If successful, this Thrust could fundamentally transform our ability to create and sustain fusion-relevant plasmas, and hasten our progress in addressing the challenges common to all fusion approaches.

Thrust 18: Achieve high-performance toroidal confinement using minimal externally applied magnetic field

Alternate configurations with magnetic field generated largely by electrical current flowing within the plasma represent potentially high payoff options for the fusion energy program. They require relatively modest external magnets, reducing engineering challenges and costs. They also have high plasma pressure, at 10-80% of the magnetic pressure. Their large ohmic heating may allow fusion ignition without complex auxiliary systems. Two of the configurations have no physical structure threading the plasma volume. These "compact torus" (CT) configurations have cylindrical plasma containment vessels that improve the accessibility and enhance the maintainability of prospective power plants.

The US is a leader in the international exploration of low external-field concepts. The reversed field pinch (RFP) looks much like a tokamak but generates most of its own magnetization. The spheromak and field-reversed configuration (FRC) are CT plasmas with distinct approaches to stability and sustainment. Magnetic fluctuation control techniques have led to important advances. These include increased energy confinement (obtaining values comparable to those in tokamak plasmas) and longer plasma lifetimes (ten times longer than without active control). Together with the tokamak and stellarator efforts, research on these configurations broadens the scientific approach to grow and validate fusion science over a wide range of plasma conditions and enhances the opportunity for scientific discovery and innovation in toroidal confinement. Also, low-field configurations exhibit processes resembling those in space and astrophysical plasmas and provide versatile laboratories for fusion, basic plasma science, and education.

Key Issues:

- Understanding confinement and stability in reactor-relevant conditions. *Will confinement continue to improve at high plasma current? How can plasma-profile and boundary control be used to achieve high performance? Can a population of energetic ions enhance stability?*

- Improving the efficiency of formation and sustainment. *Can low-frequency AC induction, waves, and neutral beams efficiently drive DC plasma current? How can formation and penetration of current sourced from outside the plasma be optimized?*

- Demonstrating the compatibility of confinement, sustainment, and plasma-boundary control. *What shaping is optimum, and how can plasma self-organization be used? What is the optimal active control system for stability? Will auxiliary heating be necessary?*

Proposed Actions:

- Develop and deploy new plasma diagnostics to measure profile information and fluctuations for the scientific goal of understanding transport and stability in low external-field devices.

- Improve and apply theoretical and computational models to analyze nonlinear effects in low-field configurations. Validate models through comparison with improved measurements.

- Study FRC stability at small ion gyroradius in a new or upgraded facility with energetic ion sources. Success will enable integrated tests of stability, confinement, and sustainment.
- Develop improved current sustainment methods for the spheromak. Small experiments will feed transformational ideas to a larger facility to test integrated confinement and sustainment.
- Extend confinement scaling and demonstrate current sustainment at high temperature in a new large-current RFP. A staged, upgradeable facility would eventually demonstrate near-burning plasma conditions with integrated plasma-boundary and magnetohydrodynamic (MHD) stability control.
- Quantify the benefits of low external field and CT geometry in system studies with updated physics and engineering information. Evaluate pulsed vs. steady-state reactor operation.

Scientific and Technical Research

Minimizing the external magnetization required to confine fusion plasma would be a major advance toward making fusion power an economical reality. This goal guides research efforts for the RFP, spheromak, and FRC magnetic configurations and motivates research that answers important scientific questions. Magnetic fluctuations arise more easily without a large externally imposed stabilizing field, and this affects both plasma stability and confinement quality. Understanding the scaling of fluctuations is crucial to predict reactor-grade performance. Efficient plasma current drive is vital, and many of the techniques are distinct from methods used for tokamaks. Control of the plasma boundary is essential for any fusion system, and optimized solutions will depend to some degree on magnetic geometry. While the integration of stability, confinement, and boundary control and other scientific challenges are particular for any magnetic configuration, the underlying physics has a large degree of commonality.

The similarities of the scientific issues for low external-field configurations help define a unified thrust; nevertheless, the configurations described in this Thrust are distinct and offer unique approaches to achieving a fusion power source. Each of them requires separate experimental work but contributes to a common physics understanding. The magnetic profile of the RFP is characteristically paramagnetic with the magnetic field orientation changing from toroidal in the core to purely poloidal near the edge. The profile therefore has large negative shear, which enhances MHD stability to interchange distortion at moderately high plasma-β (the ratio of plasma pressure to magnetic energy density). Also, many robustly damped helical modes resonate within the plasma, enhancing nonlinear stability. The toroidal geometry facilitates external control of the magnetic field at the surface of the plasma. This control has been used to achieve high confinement, and it underlies the RFP scheme for AC current drive.

Engineering requirements are further simplified for the two CT configurations. The plasma-containment volumes are basically cylindrical with no components placed along the geometric axis. Access to all plasma facing components is therefore relatively simple. The absence of external toroidal field coils allows more flexibility for designing the divertor systems that strongly influence edge plasma conditions while channeling hot plasma exhaust. However, qualitative differences exist between the spheromak and FRC. In the spheromak, the profile of magnetic field-line winding (safety factor, *q*) lies between the profiles found in tokamaks and RFPs, and can have large beneficial shear. Compatibility with divertors is well established, and formation can be achieved without a central solenoid. The FRC can be formed by fast transients and by low-frequency field rotation. Moreover, it is the highest-β confinement configuration; ideally, it has no toroidal field. Energetic large-orbit particles can compensate for the absence of stabilizing toroidal field, and FRCs have proven remarkably resilient to translation. Low total field within the confined plasma makes the FRC a leading candidate for advanced fuels, another potentially transformative area, where synchrotron radiation is prohibitive for other concepts.

Elements of the Thrust

All three of the minimal external field configurations need a range of activities to achieve their ITER-era goals. Here, we describe the activities and how they contribute to the Thrust.

- Facility upgrade and construction — whether an experimental discharge behaves qualitatively like a reactor-grade plasma depends on wave propagation relative to diffusion (Lundquist number, *S*) and on system size relative to particle gyroradius ($s = 1/\rho^*$). Achieving representative values is critical for understanding transport, current drive, and stability in all three configurations. Both parameters increase with increasing plasma current and with system size. For larger current, power supply upgrades can provide a first step for scaling information. With added flexibility, upgrades will also improve current drive and formation studies. There are limits to what can be achieved in existing facilities, however. Substantially greater capability than presently exists will be essential to demonstrate integrated solutions associated with the ITER-era goals for each of the low external-field configurations. With improved understanding from upgrades, new diagnostics, and validated modeling, new construction will provide the necessary advances through a combination of size, power, and optimization of factors such as shaping and boundary control.
- Theory and simulation — modeling efforts to understand nonlinear effects will continue to be important in all three configurations. Collaborative modeling and experimental studies will explore nonlinear interactions between macroscopic dynamics, energetic particles, and transport on scaling and β-limits. Integrated simulations will also be used as predictive tools to explore and optimize current drive systems, plasma shaping, and divertor configurations. The modeling will need 3-D macroscopic two-fluid dynamics, parallel kinetic effects from free streaming particles, and large-orbit energetic particle effects. Algorithms applied to the RFP, spheromak, and FRC presently stress different physical requirements, and modeling efforts will benefit from collaboration to make the code features compatible. Research to extend the theory of micro-turbulent transport to low-*q* conditions is important for scaling in magnetically quiescent plasmas. Analytical developments must address relatively high plasma-β and the more significant role of fluctuations with finite parallel wavelength.
- Diagnostic development — new diagnostics and analysis techniques are essential for the Thrust and should be integral parts of facility planning. Major cross cutting developments include: 1) diagnostics suitable for low-field plasma measurements, 2) structure and

techniques for integrated data analysis, and 3) turbulence and fast ion diagnostics. Accurate magnetic field and temperature profile measurements need new techniques. Turbulence and fast ion diagnostics are required in all magnetically confined plasmas, particularly for micro-turbulent transport studies. Integrated data analysis maximizes extraction of reliable information from sets of related and complementary diagnostics. It also facilitates model validation. Resources will be required for engineering and hardware, and for student training that incorporates both computational and instrumentation aspects.

- Development and application of plasma control tools — neutral beam injection and radiofrequency will greatly contribute to current drive, current and flow profile control, b-limit studies, stabilization techniques, and ion charge-exchange diagnostics. With relatively modest magnetic field and size, the well-developed and efficient positive-ion-based beam technology that operates at high power (MW) and tens of keV/nucleon energy level will be well suited for all of the Thrust's configurations. Because the RFP, spheromak, and FRC configurations have high-β, radiofrequency techniques for high density and β must be developed.
- Development for liquid walls — low particle recycling through the use of a liquid lithium limiter is a new, potentially breakthrough strategy for influencing edge plasma conditions. The sensitivity of confinement and current drive scaling to edge plasma conditions provide strong motivation for this effort. Collaboration with the LTX program at PPPL will contribute important data on low recycling physics, providing opportunities to improve next-step experiments and accelerating development of the RFP and CT concepts. Also, liquid metal walls may be particularly well suited for low external field configurations, because poloidal flows at the surface are not impeded, and smaller RFP and CT experiments can provide direct tests.
- System studies — early reactor studies for FRCs, spheromaks, and RFPs indicate that many of their engineering features would be attractive if physics challenges can be met. Since then, the evolution of CT and RFP physics, and of fusion engineering as a whole, has altered the development terrain. A critical issue is how CT and RFP reactor technology development needs differ from those of the tokamak; addressing such needs typically requires long lead times. These considerations motivate two tasks: 1) quantify the engineering, economic, and safety characteristics of CT and RFP reactors by performing modern systems studies, and 2) assess whether divertors, blankets, and shields designed for CT and RFP reactors can benefit from low external fields and relatively uniform heat and neutron fluxes.

Scale of Effort and Readiness

Research over the past decade produced significant advances in each of the three minimal external-field configurations. However, their programmatic development levels are sufficiently different as to require separate discussions regarding readiness and scale.

Reversed field pinch accomplishments include transient control of magnetic fluctuations, yielding a ten-fold improvement in global energy confinement and simultaneous keV electron and ion

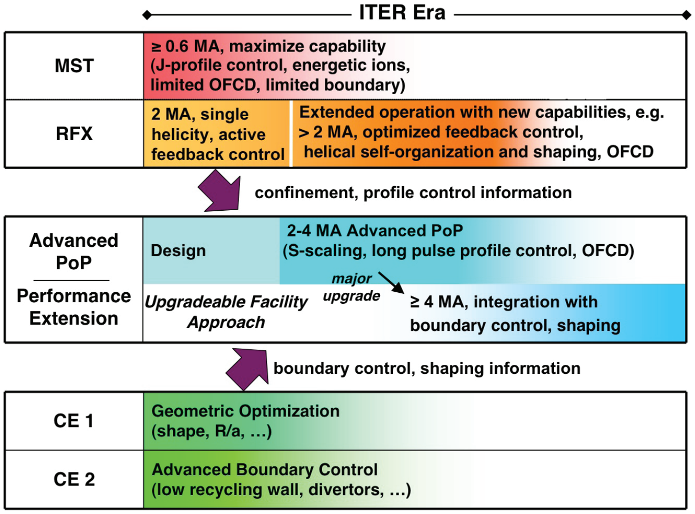

Figure 1. Upgrades and smaller experiments (CE) contribute to the advanced proof-of-principle (PoP) experiment in the RFP development path.

temperatures in MST, the only operating RFP in the US. The RFX-mod experiment in Italy and Extrap T2R in Sweden have demonstrated active control of multiple external modes to an unprecedented degree in magnetic fusion research. RFX-mod also shows an increasing likelihood of a quasi-helical state with improved confinement as the current is increased. The proposed development path shown in Figure 1 builds on these results to address concept needs.

1. Upgrading power supplies for MST can increase the maximum plasma current, possibly above 0.6 MA, decreasing ρ^* and increasing central S-value to 4×10^7. This improvement will help distinguish strong scaling of magnetic fluctuations ($\sim S^{-1/2}$, argued theoretically) from weak scaling ($\sim S^{-1/5}$, obtained at low-S in experiment and simulations). If the additional power reduces edge resistivity, AC current drive can be tested at a substantial level. Initial tests of a hybrid operating scenario using AC drive and self-similar decay will also be possible. New diagnostics, such as spectral motional Stark effect and fast Thomson scattering, will improve the accuracy of profile reconstruction and enable fluctuation studies. Installation of a 1 MW neutral beam injector will improve current profile control, allowing transport analysis in magnetically quiescent conditions. It will also provide heating to test stability limits with significant energetic-particle β.

2. Physics and engineering research for RFP boundary control will consider the technology required to spread heat loads from nearly poloidal magnetic field. Particular boundary control solutions need to consider coupled physics effects, and may benefit from recent advances in low recycling walls.

3. A study to optimize the RFP geometry will lead to more complete knowledge of fluctuations and effects that influence their scaling. An effort to minimize the deleterious effects of resonant fluctuations can begin with theoretical computations. Asymmetric shaping may increase the probability of quasi-helical states. Aspect ratio optimization, pursued by the RELAX group in Japan, will provide information on bootstrap current and mode coupling. Information from upgrades, new concept exploration (CE) experiments, and theory will influence the design of the advanced experiments described in the following task.

4. Based on the outcome of actions 1-3, a new experiment capable of demonstrating the integration of confinement, current sustainment, active control of MHD stability, and boundary control is essential to establish the scientific basis for an RFP burning plasma. This could be a two-stage, upgradeable facility that first emphasizes the integration of confinement and current sustainment at moderate plasma current. A 2 MA first stage could attain sufficiently high-S to robustly test confinement scaling, AC current drive, and the hybrid scenario. The second stage will push performance at the 4 MA level to test confinement and current-drive up to $S=10^9$ and ρ^* less than 1% with fully integrated boundary control and near-burning plasma conditions. The pulse length requirement for both stages will demand active control of resistive wall modes (RWMs). High-power plasma control using MW-level neutral beams and radiofrequency are required, and the experiment must be well-diagnosed to validate our understanding of the important physical processes.

Similar to the RFP, spheromaks have shown transient control of magnetic fluctuations leading to electron temperature of 0.5 keV and core energy confinement time of 10 ms. Continuous sustainment compatible with good confinement has not been demonstrated and is a focus of near-term activities. At present there are no large-scale experiments that can address this issue or other limits to confinement. The development path shown in Figure 2 will fill this gap, facilitate innovation to increase formation efficiency, and address technology needs.

1. Upgrades of existing and construction of new CE-level experiments will foster innovation in formation and current drive studies. With support from analytics and computation, formation experiments will investigate methods for exceeding the flux amplification and current multiplication factors obtained in past experiments. They will also improve our understanding of MHD activity excited by AC current drive and refluxing techniques for steady, quasi-steady, and pulsed operation.

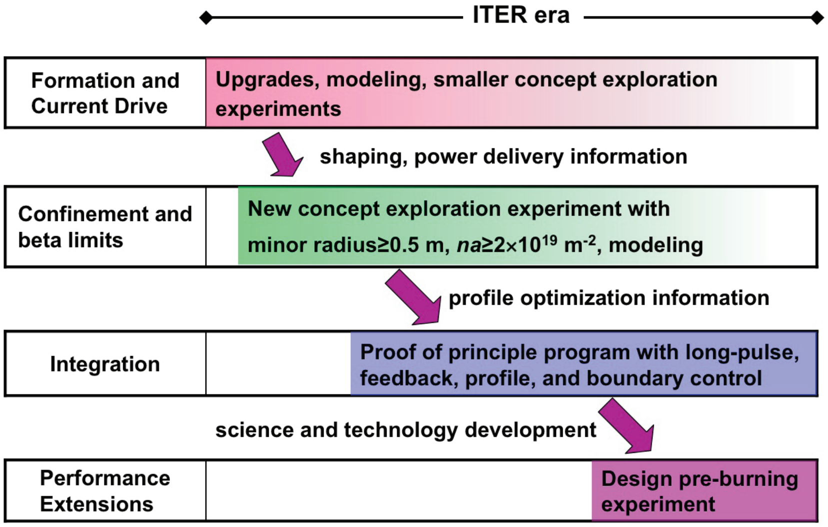

Figure 2. CE spheromaks emphasize innovation to address specific needs prior to PoP development.

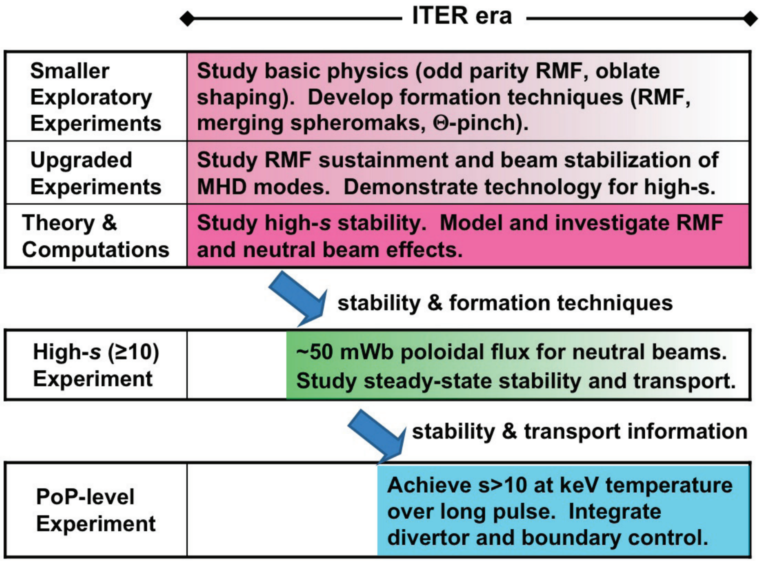

Figure 3. The FRC development path will benefit from an intermediate high-flux step.

2. Studies of confinement and β-limits require a new experiment of sufficient size to minimize charge-exchange losses. Minimizing these losses can be achieved with major and minor radii of 0.5 m or more, the product of plasma density and minor radius exceeding 2×10^{19} m^{-2}, and extensive wall conditioning. Exploring current drive will require flexible power systems that can drive plasma current in the megampere range. Current and flow profile control may rely on neutral beam and radiofrequency technologies that are tailored for low-field configurations. Feedback-controlled coils will allow time-dependent control of the flux boundary conditions.
3. With better understanding of current drive and confinement from actions 1 and 2 and from simulations, a new long-pulse (100 ms) proof-of-principle experiment will study the integration of the physics of keV plasma with boundary and profile control. It will also demonstrate the compatibility of RWM feedback with divertors and particle handling.

The FRC program path builds upon the recent advances made in formation, current drive, and theoretical understanding of kinetic stability. The development path, illustrated in Figure 3, will extend experimental research to the reactor-representative regime, where thermal ion gyro-orbits are small relative to the minor radius, and improve understanding of stability and particle transport, hence perpendicular current drive requirements.

1. Existing experimental facilities and simulations should be upgraded to realize their full potential. Increasing trapped magnetic flux will confine particles with energy nearing 10 keV so that neutral beams can be added to validate predictions of stability based on kinetic and kinetic-fluid hybrid simulations. Simulations of Rotating Magnetic Fields (RMF) can combine energetic ion modeling with boundary conditions for time-dependent fields from antennas. This will help in understanding RMF effects on stability and particle confinement, and to optimize current drive.

2. With information from action 1, construction of a new FRC experiment will be central to understanding scaling to high-*s* conditions. Basic plasma parameters impose clear requirements for achieving $s \geq 10$. For low collisionality, a variable separatrix radius up to 1 m and poloidal flux of 20-50 mWb will be required to confine 10-20 keV energetic ions in the desired thermal-s range. High current (~10-50 Amp), medium energy (20-40 keV) neutral beams will allow a significant fraction of the current to be carried by energetic particles. Flexibility with respect to shaping is also important for addressing stability. Transport studies will be combined with the high-*s* studies since cross-field transport in FRCs is related to the ratio of electron drift velocity to ion thermal velocity (γ_d). Experiments and calculations show anomalous resistivity decreasing sharply as γ_d is reduced below unity. In FRCs $\gamma_d \sim 2.3/s$, so high-s FRCs will naturally have $\gamma_d < 1$. High-energy axis encircling ions will also affect transport, so studies of high-*s* stability and transport are synergistic. Density fluctuation measurements will require laser scattering diagnostics.

3. The outcome of actions 1 and 2 will be used for a proof-of-principle scale experiment that will integrate high-β operation with boundary plasma control.

Integration of Elements

Development of the scientific and technological basis for economical fusion power through minimal applied field unifies the elements of this Thrust. The configurations cover a space of parameters that extends from large paramagnetism to extreme diamagnetism, over varying degrees of magnetic shear, and from moderate to large β. While device-specific development paths should be pursued, the common activities listed earlier will foster greater collaboration. The three configurations need similar diagnostic developments, and profile and boundary control. Collaborations will increase the versatility of simulation capabilities and the reliability of numerical predictions. Finally, system studies will help quantify engineering tradeoffs.

Other Scientific Benefits

The study of low external field configurations has strong ties to the areas of space, solar, and astrophysical plasmas. Magnetic self-organization in RFPs and spheromaks is related to dynamo, which is considered important for the generation of interstellar and intergalactic fields. Current collimation and the MHD stability of stellar and astrophysical jets are analogous to basic properties of the RFP and spheromak. Magnetic reconnection and associated ion heating are also important in these configurations, as they are in solar and space plasmas.

With their robust stability, FRCs with large ion gyro-radii (low-*s*) have potential application beyond steady confinement schemes. Additional information gained from existing and planned facilities of this Thrust can be useful for high energy density laboratory plasma (HEDLP) programs such as magneto-inertial fusion. It will also influence the development of plasma thrusters for space propulsion.

Over the last two decades, low external field research in the US has largely benefitted from and contributed to fusion science efforts at our major research universities. Its exploratory nature fosters creative thinking and motivates scientific discovery in fusion science. It will continue to have important educational value throughout the ITER era.

Connections to Other Thrusts

- For the RFP, three-dimensional shaping for improving confinement by externally inducing quasi-helical states is an activity in Thrust 17. Aspect ratio optimization to enhance bootstrap current and to influence mode coupling is related to Thrust 16.
- The integrated data analysis described earlier is critical for experiments such as ITER, where diagnostic access is extremely restricted and the need for validated data is extremely high. Turbulence and fast ion diagnostics are required for measurement of transport mechanisms in all magnetically confined plasmas. (Thrusts 1 and 6)
- Predictive modeling that integrates microturbulence with macroscopic dynamics for low-field devices requires strong electromagnetic effects, including changing magnetic topology. It will rely on sufficiently general development from Thrust 6, and validation requires diagnostic capabilities that measure turbulence in low-field conditions. In turn, efforts for Thrust 18 will broaden the parameter space of predictive simulation capability.
- Low recycling wall development is a potentially breakthrough area for all three low-field configurations and is an important activity in Thrust 11.
- CT injection of fuel and momentum is a potential method of controlling the profiles of density, fusion heating, and rotation for ITER and DEMO. (Thrust 13)
- Solenoid-free formation and current drive are strengths of the CT program and will continue to contribute to low aspect ratio tokamak development. (Thrust 16)

APPENDIX A: ACRONYMS AND ABBREVIATIONS

2-D	Two-dimensional
3-D	Three-dimensional
4-D	Four-dimensional

AC	Alternating Current
AE	Alfvén Eigenmode
ALPS	Advanced Liquid Plasma-facing Surface Program (DOE Program)
APEX	Advanced Power Extraction Program (DOE Program)
APS	American Physical Society
APS/DPP	American Physical Society/Division of Plasma Physics
ARIES	Advanced Reactor Innovation Evaluation Studies
ARIES-AT	ARIES-Advanced Tokamak Design
ARIES-RS	ARIES Reverse Shear Design
ARIES-CS	ARIES Compact Stellarator Design
ASDEX	A tokamak experiment at Max-Planck Institute, Germany
ASME	American Society of Mechanical Engineers
AT	Advanced Tokamak
AUG	ASDEX-Upgrade, a tokamak in Germany
AWCD	Alfvén Wave Current Drive
AWH	Alfvén Wave Heating

B	Magnetic field
B2-Eirene	A two-dimensional plasma-fluid transport code
BES	Office of Basic Energy Sciences, Department of Energy
BES	Beam Emission Spectroscopy
BOUT	An edge plasma turbulence simulation code
BR	Breeding Ratio
BSCCO-2212	A high-temperature superconductor material

CAD	Computer-aided Design
CAE	Compressional Alfvén Eigenmode
CE	Concept Exploration
CER	Charge Exchange Recombination (spectroscopy diagnostic)
CFC	Carbon Fiber Composite
CHI	Coaxial Helicity Injection
CICC	Cable-in-conduit-conductor
C-Mod	(or Alcator C-Mod) A tokamak at the MIT Plasma Science and Fusion Center
Colorado FRC	Colorado Field-Reversed Configuration
CPES	Center for Plasma Edge Simulation
CT	Compact Torus
CTF	Component Test Facility
CTH	A compact stellarator-tokamak hybrid at Auburn University in Alabama
CTX	A spheromak experiment (operations ended) at LANL
CW	Continuous Wave

DALFTI	An edge plasma turbulence simulation code
DC	Direct Current
DCLL	Dual Coolant Lead Lithium
D-D	Deuterium-Deuterium (plasma-fuel interaction)
DEBS	A simulation code
DEMO	Demonstration fusion power plant

ACRONYMS AND ABBREVIATIONS (continued)

DIII-D	A tokamak at GA
DOE	Department of Energy
D-T	Deuterium-Tritium (plasma-fuel interaction)
EAST	Experimental Advanced Superconducting Tokamak in China
EBW	Electron Bernstein Wave
ECCD	Electron Cyclotron Current Drive
ECDC	Electron Cyclotron Discharge Cleaning
ECE	Electron Cyclotron Emission
ECH	Electron Cyclotron Heating
ECRF	Electron Cyclotron Range of Frequencies
EDA	Enhanced Dα Mode
EDGE2D	A two-dimensional plasma-fluid transport code
ELM	Edge Localized Mode
EP	Energetic Particle
EPM	Energetic Particle Mode
EPRI	Electric Power Research Institute
ESEL	An edge plasma turbulence simulation code
ETG	Electron Temperature Gradient
EU	European Union
eV	Electron Volt
Extrap-T2R	Reversed field pinch experiment at the Royal Institute of Technology, Sweden
FACETS	Computational modeling framework
FCI	Flow Channel Insert
FESAC	Fusion Energy Sciences Advisory Committee
FLR	Field-line Resonance
FNSF	Fusion Nuclear Science Facility
Fr	Failure rate
FRC	Field-reversed Configuration
FRX-L	A field-reversed configuration plasma source at LANL
FSP	Fusion Simulation Program
FTE	Full Time Equivalent
FTU	Frascati Tokamak Upgrade, a tokamak in Italy
FZK	Research Center in Karlsruhe, Germany
GA	General Atomics, California
GAE	Global Alfvén Eigenmode
GENE	A simulation code
GHz	Gigahertz
GPI	Gas Puff Imaging
GW	Gigawatt
HDH	High-density H-mode
HHFW	High Harmonic Fast Wave
HI	(Magnetic) Helicity Injection
HIBP	Heavy Ion Bean Probe
HIT-SI	Steady Inductive Helicity Injected Torus at the University of Washington, Seattle
H-Mode	High-confinement Mode
HSX	Helically Symmetric Experiment, a stellarator at the University of Wisconsin-Madison
HTS	High-temperature Superconductors
HYM	A numerical simulation code

ACRONYMS AND ABBREVIATIONS (continued)

IBW	Ion Bernstein Wave
ICRF	Ion Cyclotron Range of Frequencies
ICW	Ion Cyclotron Wave
IFMIF	International Fusion Materials Irradiation Facility
INL	Idaho National Laboratory, Idaho
ITB	Internal Transport Barrier
ITER	International burning plasma experiment being built in Cadarache, France
ITG	Ion-temperature Gradient Mode
ITPA	International Tokamak Physics Activity

JAEA	Japan Atomic Energy Agency
JAERI	Japan Atomic Energy Research Institute
JET	European tokamak sited in the UK
JT-60SA	Japanese Tokamak-Super Advanced (under construction) at JAERI
JT-60U	Japanese Tokamak-Upgrade at JAERI

kA	Kiloampere
KBM	Kinetic Ballooning Mode
keV	Kiloelectron Volt
KSTAR	Korea Superconducting Tokamak Advanced Research device in South Korea
kV	Kilovolt

low-A	Low aspect ratio
LANL	Los Alamos National Laboratory, New Mexico
LANL-DRX	A spheromak experiment at LANL
LDX	Levitated Dipole Experiment at MIT
LH	Lower Hybrid
L-H	Low-to-high-confinement Mode (transition)
LHCD	Lower Hybrid Current Drive
LHD	Large Helical Device, a stellarator experiment in Japan
LHRF	Lower Hybrid Radiofrequency
Li	Lithium
LLNL	Lawrence Livermore National Laboratory, California
L-mode	Low-confinement Mode
LSX	Large-*S* Experiment, a field-reversed configuration experiment at the University of Washington, Seattle
LTS	Low-temperature Superconductor
LTX	Lithium Tokamak Experiment at PPPL

MA	Megampere
MARS	A computational code
MAST	Mega Ampere Spherical Tokamak at Culham Laboratory, UK
MAST-U	Mega Ampere Spherical Tokamak-Upgrade at Culham Laboratory, UK
MCCD	Mode Conversion Current Drive
MCH	Mode Conversion Heating
MELCOR	A safety analysis computer code
MeV	Mega Electron Volt
MFE	Magnetic Fusion Energy
MHD	Magnetohydrodynamics
MIT	Massachusetts Institute of Technology, Massachusetts
MHz	Megahertz

ACRONYMS AND ABBREVIATIONS (continued)

MJ	Mega Joule
MRX	Magnetic Reconnection Experiment at PPPL
MST	Madison Symmetric Torus at the University of Wisconsin-Madison
MTBF	Mean Time Between Failures
MTTR	Mean Time to Repair or Replace
mV	Millivolt
MW	Megawatt
mWb	Milli-weber, a unit of magnetic flux

NB	Neutral Beam
NBCD	Neutral Beam Current Drive
NBI	Neutral Beam Injection
NBICD	Neutral Beam Injection Current Drive
NCSX	National Compact Stellarator Experiment (cancelled) at PPPL
NIMROD	A simulation code
NNBI	Negative-ion-based Neutral-beam Injection
NOVA-K	A simulation program
NRC	National Research Council
NRC	Nuclear Regulatory Commission
NSTX	National Spherical Torus Experiment at PPPL
NSTX-U	National Spherical Torus Experiment-Upgrade at PPPL
NTM	Neoclassical Tearing Mode

OFCD	Oscillating Field Current Drive
OFES	Office of Fusion Energy Sciences
OH	Ohmic Heating
ORBIT	A simulation code
ORNL	Oak Ridge National Laboratory, Tennessee
OSM-Eirene	A two-dimensional plasma-fluid transport code

PBX	A spheromak experiment at Woodruff Scientific, Seattle, Washington
PE	Performance Extension
PEGASUS	A spherical torus experiment at the University of Wisconsin-Madison
PFC	Plasma Facing Component
PFRC	A simulation code
PGO	Priorities, Gaps, and Opportunities (FESAC Report)
PHD	A reversed field pinch experiment at the University of Washington, Seattle
PIC	Particle-in-Cell Code
PMI	Plasma-material Interaction
PoP	Proof of Principle
ppm	parts per million
PPPL	Princeton Plasma Physics Laboratory, Princeton University, New Jersey
PSFC	Plasma Science and Fusion Center, MIT, Massachusetts
PSI	Plasma-surface Interaction
PV-Rotomak	A reversed field pinch experiment at Prairie View A&M, Texas
P-V-T	Pressure Volume Temperature
PWI	Plasma-wall Interaction

q	Safety factor for stability
Q	The ratio between fusion power produced and heating power supplied
QA	Quasi-axisymmetry
QH	Quasi-helical Symmetry

ACRONYMS AND ABBREVIATIONS (continued)

QP	Quasi-poloidal Symmetry
QS	Quasi-symmetric
QSH	Quasi-single Helicity State
QUEST	A spherical torus experiment at Kyushu University, Japan

R&D	Research and Development
RAFMS	Reduced Activation Ferritic Martinsitic Steel
RAMI	Reliability, Availability, Maintainability, and Inspectability
ReBCO	Rare Earth-Barium-Copper-Oxide
RELAX	A reversed field pinch experiment at Kyoto Institute of Technology, Japan
ReNeW	Research Needs Workshop (Magnetic Fusion Energy Sciences)
RF	Radiofrequency
RFP	Reversed Field Pinch
RFX	A reversed field pinch experiment in Padova, Italy
RMF	Rotating Magnetic Field
rMHD	Resistive Magnetohydrodynamics
RMP	Resonant Magnetic Perturbation
RSX	Reconnection Spheromak Experiment at LANL
RWM	Resistive Wall Mode

SciDAC	Scientific Discovery through Advanced Computing
SC	Superconducting
SDC	Super Dense Core
SiC	Silicon Carbide
SOL	Scrape-off Layer
SRNL	Savannah River National Laboratory, South Carolina
SSPX	Sustained Spheromak Physics Experiment at LLNL
SST-1	Steady State Tokamak Experiment in India
SSTR	Steady State Tokamak Reactor
SSX	Swarthmore Spheromak Experiment at Swarthmore College, Pennsylvania
ST	Spherical Torus
ST-CTF	Spherical Torus-Component Test Facility
START	Small Tight Aspect Ratio Tokamak (operations ended) at Culham Laboratory, UK

T-11M	A tokamak device in Russia
T2R	Extrap-T2R Experiment at the Royal Institute of Technology, Sweden
TAE	Toroidal Alfvén Eigenmode
TAP	Toroidal Alternates Panel
TBM	Test Blanket Module
TCSU	A field-reversed configuration experiment at the University of Washington-Seattle
TF	Toroidal Field
TFTR	Tokamak Fusion Test Reactor (decommissioned) at PPPL
TITAN	A reversed-field pinch reactor study
Tore Supra	A superconducting tokamak experiment in France
TORIC	A simulation code
TRANSP	A time-dependent tokamak transport analysis code
TRL	Technical Readiness Level
TSTA	Tritium Systems Test Assembly at LANL

UCLA	University of California at Los Angeles
UCSD	University of California at San Diego
UEDGE	A two-dimensional plasma-fluid transport code

ACRONYMS AND ABBREVIATIONS (continued)

USIPO	US ITER Project Office
VDE	Vertical Displacement Event
VNS	Volumetric Neutron Source
VUV	Vacuum Ultraviolet
W	Tungsten
W7-AS	Wendelstein-7 Advanced Stellarator, a stellarator experiment in Germany
W7-X	Wendelstein-7X, a stellarator being built in Germany
W-fuzz	Tungsten fuzz
YBCO	Yttrium Barium Copper Oxide, a high-temperature superconductor material
Z	Atomic number
$\mathbf{Z_{eff}}$	Effective atomic number, a measure of impurity content

APPENDIX B: WORKSHOP SCHEDULE

	Monday, June 8	**Tuesday, June 9**	**Wednesday, June 10**	**Thursday, June 11**	**Friday, June 12**
8:30 AM	Plenary (P): welcome and logistics	**P: Dr. Ed Synakowski, AD for Fusion: Context for ReNeW**	**B: Thrust focus discussion.** 3 parallel sessions: 4, 9, 7	P: Summary of thrust revisions	P: Wrap-up for OFES (all invited)
9:00 AM	**P: Dr. Dehmer: Perspectives from Office of Science**				
9:30 AM		Thrust 6	Thrust focus discussion: 5, 10, 18		
10:00 AM	Break	Break		Break	Break
10:30 AM	P: Theme 1 Chapter summary and discussion (20 + 10)	Thrust 7	Break	P: Begin final discussions of Thrust content	Excom meets for wrap-up
11:00 AM	P: Theme 2 Chapter summary and discussion	Thrust 8	Thrust focus discussion: 2, 11, 14		
11:30 AM	P: Theme 3 Chapter summary and discussion	Thrust 9			
12:00 PM	Break	Break	Break	Break	Adjourn
12:30 PM	Break	Break	Break	Break	
1:00 PM	P: Theme 4 Chapter summary and discussion	Thrust 10	Thrust focus discussion: 8, 1, 15	P: Continue thrust content finalization	
1:30 PM	P: Theme 5 Chapter summary and discussion	Thrust 11			
2:00 PM	Breakout (B): Theme chapter improvements	Thrust 12	Thrust focus discussion: 12, 3, 17		
2:30 PM		Thrust 13			
3:00 PM	**P: Thrust summary and discussion.** 20 + 10 for each Thrust. Thrust 1	Thrust 14	Break	P: Discuss executive summary	
3:30 PM	Break	Break	Thrust focus discussion: 16, 6, 13	Break	
4:00 PM	Thrust 2	P: Continue thrust summary and discussion. Thrust 15			
4:30 PM	Thrust 3	Thrust 16	Free time	B: Writing chores	
5:00 PM	Thrust 4	Thrust 17			
5:30 PM	Thrust 5	Thrust 18	P: thrust discussion		
6:00 PM	**Reception**	Excom meeting			

APPENDIX C: THEME WORKSHOPS

THEME WORKSHOPS

To prepare for the ReNeW workshop, five Theme workshops were held in March of 2009. The purpose of these workshops was to solicit input from the wider fusion energy sciences community to identify and refine the scientific research requirements of each Theme and formulate preliminary research thrusts.

A list of these workshops, along with links to the websites where the workshop presentations are posted, is included below:

THEME 1 & 2 JOINT WORKSHOP

March 23-27, 2009
General Atomics, San Diego
http://fusion.gat.com/global/Renewt12

THEME 3 WORKSHOP

March 4-6, 2009
UCLA
http://www.fusion.ucla.edu/FNST/Renew_Presentations/

THEME 4 WORKSHOP

March 2-4, 2009
UCLA
http://www.fusion.ucla.edu/FNST/Renew_Presentations/

THEME 5 WORKSHOP

March 16-19, 2009
Princeton Plasma Physics Laboratory
http://www.pppl.gov/conferences/ReNeW/T5Workshop/index.html

APPENDIX D: WHITE PAPERS

COMMUNITY WHITE PAPERS

During the period leading up to the June workshop in Bethesda, the magnetic fusion energy sciences community had the opportunity to submit brief descriptions of proposed research thrusts or other input addressing scientific and technical issues or concerns of relevance to the scope of the ReNeW. Submissions were welcome from all members of the fusion community, regardless of whether they were formally participating in ReNeW. In addition, the authors of these white papers were offered the opportunity to present their input during the five Theme workshops held in March of 2009. More than 250 white papers were submitted in response to this request, and they were posted on the public page of the ReNeW website. Considering their importance in informing and formulating the thrusts that emerged from the ReNeW, these white papers will be maintained online for at least five years.

A list of these white papers arranged by Theme is included below. Links to the white papers can be found at: http://burningplasma.org/web/renew_whitepapers.html

THEME 1: BURNING PLASMAS IN ITER

1. G. Bateman and A.H. Kritz, *Improvements needed for Predictive Integrated Modeling of Plasma Rotation, Density and Temperature Profiles*
2. L.R. Baylor, *Fusion Burn Control with Isotopic Fueling*
3. R.L. Boivin, J. C. DeBoo, and R.K. Fisher, *A Fusion Development Facility to Develop, Field, and Test Diagnostic Solutions for DEMO*
4. R.J. Buttery, *The Need for a Fusion Science Integration Experiment in the US*
5. V.S. Chan, *Creating a Self-Heated Plasma*
6. T.E. Evans, *Edge Localized Mode and pedestal control using resonant magnetic perturbation coils*
7. J. Ferron, *Global parameter control in ITER and a steady-sate DEMO*
8. J. Ferron, *Profile control in a steady-state advanced tokamak*
9. D.A. Gates, *Integrated stability control*
10. D.A. Gates, *Research gaps for reactor startup sequencing in the ITER era*
11. J. Harris, *Perspectives for operation of stellarators as fusion reactors*
12. J.W. Hughes, D.R. Mikkelsen, C.C. Petty, J.E. Rice, W.L. Rowan, and P.W. Terry, *Power thresholds and other requirements for H-mode access in ITER scenarios*
13. J.W. Hughes, D.R. Mikkelsen, C.C. Petty, J.E. Rice, W.L. Rowan and P.W. Terry, *Physics-based prediction of edge pedestal profiles under reactor-like conditions*

14. D.L. Humphreys, *Active Realtime Control Issues and Role of a Fusion Development Facility*

15. A.H. Kritz, R. Budny, and G. Bateman, *Open Issues for Integrated Modeling*

16. R.J. La Haye, *Tearing Mode Avoidance and Stabilization*

17. K.C. Lee, *Modeling of Turbulence Diffusion and H-mode Transition including Poloidal Momentum Exchange from Ion-Neutral Collisions*

18. M.A. Makowski, D.D. Ryutov, A. Hyatt, T.H. Osborne, and M.V. Umansky, *Control of a Snowflake divertor*

19. E. Mazzucato, D. R. Smith, and K.L. Wong, *Systematic Investigation of Electron Transport in NSTX*

20. D. Meade, *Scientific Issues and Gaps for High-Performance Steady-State Burning-Plasmas*

21. D.R. Mikkelsen, *Transport in 'transient' conditions: current ramp-up & ramp-down, between sawtooth crashes*

22. R. Moyer, T.E. Evans, R. Groebner, H. Reimerdes, and P. Snyder, *Pedestal Optimization for Confinement and ELM Control*

23. R. Nazikian and M. A. Van Zeeland, *Alpha particle transport in high temperature steady state reactor regimes*

24. R.E. Nygren, *Future Plasma Facing Components (PFCs) & In-vessel Components (IVCs): Strengthened Sustained and Integrated Approach for Modeling and Testing HHFCs*

25. K. Owens, *Burning Plasma Control*

26. T.W. Petrie, *Some issues related to core and divertor control for DEMO*

27. T.W. Petrie, *Some issues related to core and divertor control for ITER*

28. C.C. Petty, *Effect of Toroidal Field Ripple on Edge Confinement*

29. C.C. Petty, *Extrapolating Transport to the Burning Plasma Regime*

30. H. Reimerdes, J.W. Berkery, A.M. Garofalo, Y. In, R.J. La Haye, M. Okabayashi and S.A. Sabbagh, *Resistive Wall Mode Stabilization*

31. J.E. Rice*, Driven and intrinsic rotation, rotation sinks and momentum transport*

32. J.E. Rice, *ITB formation from velocity and magnetic shear*

33. W.L. Rowan, J.W. Hughes, D.R. Mikkelsen, C.C. Petty, J. E. Rice, and P.W. Terry *Particle & Impurity Transport and Fuelling*

34. R.D. Stambaugh, V.S. Chan, A.M. Garofalo, et al., *Research Thrusts Made Possible by a Fusion Development Facility*

35. R.D. Stambaugh, V.S. Chan, A.M. Garofalo, et al., *Research Thrusts Made Possible by a Fusion Development Facility — executive summary*

36. E.J. Strait, J.C. Wesley, M.J. Schaffer, and M.A. Van Zeeland, *Off-Normal Events in a Fusion Development Facility*

37. P.W. Terry, J.W. Hughes, D.R. Mikkelsen, C.C. Petty, J.E. Rice, and W.L. Rowan, *Toward a Validated Understanding of Core Thermal Transport*

38. A. Turnbull, *Control of the Plasma and the Power Flow in a Reactor*

39. A. Turnbull, *Disruptions And ELMs: Questions That Need Answers Before DEMO*

40. A. Turnbull, *Evaluation of Technology Readiness for Physics-Oriented Issues*

41. A. Turnbull, *Technology Readiness For Control of the Plasma and Power Flow in a Reactor*

42. M.A. Van Zeeland, *Fast Ion Driven Instabilities Generated by Alpha Particles in Future Burning Plasma Experiments*

43. M. Walker and E.J. Strait, *Off-Normal Event Control*

44. M. Walker, *Control Algorithms and Approaches*

45. T.L. Weaver, D.E. Turner and J.A. Ramsey, *Multivariate Control in Fusion Energy: Need for a Virtual Fusion Controls Development Laboratory*

46. T.L. Weaver, *Regulation and Licensing: Issues for Control Systems*

47. T.L. Weaver, *Reliability, Maintainability, and Testability: Issues for Control Systems*

48. J.C. Wesley, *Disruptions – A Personal View*

49. S.M. Wolfe, *Burn Control and Thermal Stability*

50. G.A. Wurden, *The Real Mission of ITER*

THEME 2: CREATING PREDICTABLE HIGH-PERFORMANCE STEADY-STATE PLASMAS

1. L.R. Baylor, *Fusion Burn Control with Isotopic Fueling*
2. R.L. Boivin, J.C. DeBoo, and R.K. Fisher, *A Fusion Development Facility to Develop, Field, and Test Diagnostic Solutions for DEMO*
3. A. Boozer, G.H. Neilson, and M. Zarnstorff, *Non-Axisymmetric Shaping as a Research Thrust*
4. R.J. Buttery, *Disruptions – Research Needs*
5. R.J. Buttery, *The Need for a Fusion Science Integration Experiment in the US*
6. P. Cadaret, *Advanced Data Compression Methods (ADCM)*
7. P. Cadaret, *Complex Model Acceleration Using Neural Networks*
8. P. Cadaret, *Empirically Trained Diagnostic Systems (ETDS)*
9. V.S. Chan, *Validated Theory and Predictive Modeling*
10. D.A. D'Ippolito and J.R. Myra, *ICRF-Edge and Surface Interactions*
11. Edge Coordinating Committee (ECC), *Integrated Edge-Plasma and Plasma-Wall Interaction Research*
12. L. El-Guebaly and R. Raffray, *Impact of Tritium Breeding on Design Implications to Withstand Disruptions and Runaway Electrons*
13. L. El-Guebaly, *Need for High Tritium Burn-up Fraction in Plasma to Relax Tritium Breeding Requirement for Demo and Fusion Power Plants*
14. T.E. Evans, *Edge Localized Mode and pedestal control using resonant magnetic perturbation coils*
15. J. Ferron, *Global parameter control in ITER and a steady-sate DEMO: a white paper for ReNeW*
16. J. Ferron, *Profile control in a steady-state advanced tokamak: a white paper for ReNeW*
17. A.M. Garofalo, *Integration of High Performance, Steady-State Burning Plasmas*
18. D.A. Gates, *Integrated stability control*
19. D.A. Gates, *Research gaps for reactor startup sequencing in the ITER era*
20. R.J. Goldston, *The Role of a Long-Pulse, High-Heat-Flux, Hot-Walls Confinement Experiment in the Study of Plasma-Wall Interactions for CTF and Demo*
21. R.J. Goldston, *The Role of a Long-Pulse, High-Heat-Flux, Hot-Walls Confinement Experiment in the Development of Plasma Facing Components for CTF and Demo*
22. R.J. Goldston, *Requirements for a Confinement Device with a Goal to Develop Tritium Breeding Blanket Modules, Based on FESAC Fusion Development Path Plan*

23. L.R. Grisham, *Neutral Beam Development Plans and Needs for ITER*
24. J. Harris, *Perspectives for operation of stellarators as fusion reactors*
25. D.L. Humphreys, *Active Realtime Control Issues and Role of a Fusion Development Facility*
26. Y. In, M. Okabayashi, E.J. Strait, H. Reimerdes, and R.J. La Haye, *Holistic approach against performance-limiting instabilities in steady state plasmas*,
27. T.P. Intrator and R.P Majeski, *Radio frequency sustainment and current drive for Compact Tori*
28. M. Kotschenreuther, S.M. Mahajan, P. Valanju, J.M. Canik, A.M. Garofalo, B. LaBombard, and R. Maingi, *Severe divertor issues on next step devices, and validating the Super-X divertor as a promising solution*
29. A.H. Kritz, R. Budny, and G. Bateman, *Open Issues for Integrated Modeling*
30. R.J. La Haye, *Issue: Tearing Mode Avoidance and Stabilization*
31. K.C. Lee, *Modeling of Turbulence Diffusion and H-mode Transition including Poloidal Momentum Exchange from Ion-Neutral Collisions*
32. J. Leuer and D. Humphreys, *Solenoidless Startup Research Thrusts for Tokamaks*
33. M.A. Makowski, D.D. Ryutov, A. Hyatt, T.H. Osborne, and M.V. Umansky, *White Paper: Control of a Snowflake divertor*
34. J. Manickam, *Development of a Stability Dashboard for Tokamaks*
35. D. Meade, *Scientific Issues and Gaps for High-Performance Steady-State Burning-Plasmas*
36. J. Menard and R. J. Goldston, *Contributions of NHTX to high-performance steady-state plasma development*
37. J. V. Minervini, L. Bromberg, D. C. Larbalestier, M. Gouge, B. E. Nelson, R. J. Thome, and GianLuca Sabbi, *US Fusion Program Issues and Requirements for Superconducting Magnets Research*
38. R. Moyer, T. E. Evans, R. Groebner, H. Reimerdes, and P. Snyder, *Pedestal Optimization for Confinement and ELM Control*
39. J.R. Myra and D.A. D'Ippolito, *Comments on Verification and Validation in Edge Research*
40. R.E. Nygren, *Future Plasma Facing Components (PFCs) & In-vessel Components (IVCs): Strengthened Sustained and Integrated Approach for Modeling and Testing HHFCs*
41. R.E. Nygren, *In-vessel Engineering Instrumentation for Future Fusion Devices*
42. K. Owens, *Burning Plasma Control*

43. T.W. Petrie, *White Paper: Some issues related to core and divertor control for DEMO*

44. T.W. Petrie, *White Paper: Some issues related to core and divertor control for ITER*

45. C.C. Petty, *Hybrid Regime as an Alternative Scenario for High-Beta, Steady-State Plasmas*

46. C.K. Phillips, *Comments on research requirements/strategies for addressing "Plasma Modification by Auxiliary Systems" with RF waves*

47. P. Politzer, *Integration of Issues for Control, Research Needs, and a Possible Thrust*

48. N. Pomphrey, A. Reiman, and A.H. Boozer, *Benefits of moderate 3D fields in Tokamaks - NHTX as an example*

49. R. Prater and V.S. Chan, *A Fusion Development Facility to Test Heating and Current Drive Systems for DEMO*

50. R. Prater, *Provide basis for tokamak operation with dominant electron heating*

51. R. Raman, T.R. Jarboe, and H.W. Kugel, *Research Thrust on Advanced Plasma Fueling*

52. R. Raman, T.R. Jarboe, B.A. Nelson, D. Mueller, and M.G. Bell, *Research Thrust on Plasma Startup & Ramp-up*

53. A. Reiman, *Stellarator* β *Limits*

54. A. Reiman, *Development of a Predictive Simulation Capability for Three-Dimensional Configurations*

55. H. Reimerdes, J.W. Berkery, A.M. Garofalo, Y. In, R.J. La Haye, M. Okabayashi and S.A. Sabbagh, *Resistive Wall Mode Stabilization*

56. T. D. Rognlien, *Prediction of PFC plasma fluxes by improved edge / scrape-off-layer simulations*

57. S.A. Sabbagh, J.W. Berkery, J.M. Bialek, D.A. Gates, S.P. Gerhardt, R. J. Goldston, C.C. Hegna, O. Katsuro-Hopkins, R. J. La Haye, J.E. Menard, Y. K.-M. Peng, H. Reimerdes, and E.J. Strait, *Cross-cutting Tokamak/ST Research Needs for Stability and Steady-State Control*

58. M. A. Shapiro, J. R. Sirigiri, R. J. Temkin, D. A. Rasmussen, R. L. Ives, J. Lohr and C. Moeller, *ECH for DEMO*

59. D.A. Spong, *Fast ion stability and wall heat loading in stellarators*

60. R.D. Stambaugh, V. S. Chan, A.M. Garofalo, et al., *Research Thrusts Made Possible by a Fusion Development Facility*

61. R.D. Stambaugh, V. S. Chan, A.M. Garofalo, et al., *Research Thrusts Made Possible by a Fusion Development Facility—executive summary*

62. E.J. Strait, D.L. Humphreys, J.R. Ferron, R.J. La Haye, and M.L. Walker, *Establishing the Basis for Reliable Steady-State Fusion Power*

63. E.J. Strait, J.C. Wesley, M.J. Schaffer, and M.A. Van Zeeland, *Off-Normal Events in a Fusion Development Facility*

64. D. Stutman, M. Finkenthal, K. Tritz, L. Delgado-Aparicio, *Development of X-ray sensors and optical light extractors for DEMO operation and control*

65. R.J. Temkin, M.A. Shapiro, J.R. Sirigiri, and C.P. Moeller, *RF Launchers that Survive in the Fusion Reactor Environment*

66. A. Turnbull, *Control of the Plasma and the Power Flow in a Reactor*

67. A. Turnbull, *Disruptions and ELMs: Questions that Need Answers Before DEMO*

68. A. Turnbull, *Evaluation of Technology Readiness for Physics-Oriented Issues*

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71. M. Walker, *Control Algorithms and Approaches*

72. T.L. Weaver, D.E. Turner, and J.A. Ramsey, *Multivariate Control in Fusion Energy: Need for a Virtual Fusion Controls Development Laboratory*

73. T.L. Weaver, *Regulation and Licensing: Issues for Control Systems*

74. T.L. Weaver, Reliability*, Maintainability, and Testability: Issues for Control Systems*

75. J. Wesley, *Disruptions – A Personal View*

76. D. Whyte, *An Energy Sustainment Science Mission*

77. D. Whyte, P. Bonoli, L. Bromberg, M. Greenwald, A. Hubbard, B. Lipschultz, and J. Terry, *Contributions of a sustained high heat flux facility to "Creating predictable, high performance steady state plasmas"*

78. D. Whyte, *White paper on alpha losses and high-Z metal PFC surface damage*

79. S.M. Wolfe, *Burn Control and Thermal Stability*

80. C.P.C. Wong, C.C. Baker, and S. L. Milora, *US Fusion DEMO Assessment*

81. S.J. Wukitch, *ICRF Actuator for DEMO*

82. G.A. Wurden, *The Real Mission of ITER*

THEME 3: TAMING THE PLASMA MATERIAL INTERFACE

1. J.P. Allain, *Development and qualification of innovative advanced refractory alloys for steady-state burning plasma-wall interface materials*
2. L.R. Baylor, J. M. Canik, J.B.O. Caughman, S. Diem, R.H. Goulding, D.L. Hillis, P.S. Krstic, F.W. Meyer, and Y.K.- M. Peng, *Research Thrust to Address PMI Knowledge Gaps for DEMO*
3. J.N. Brooks, J.P. Allain, and T.D. Rognlien, *Thrust for Enhancing Modeling & Predictive Computations for Plasma/Material Interactions*
4. R.J. Buttery, *The Need for a Fusion Science Integration Experiment in the US*
5. R.W. Callis, *Taming the Plasma Material Interface, RF Antennas, Launching Structures, and Other Internal Components*
6. J.M. Canik, R. Maingi, R.J. Goldston, and J. Menard, *Simulating the Demo Edge Plasma in a Compact High Heat Flux Experiment*
7. J.B.O. Caughman, D.A. Rasmussen, L.A. Berry, R.H. Goulding, D.L. Hillis, P.M. Ryan, L. Snead, R.I. Pinkster, J.C. Hosea and J.R. Wilson, *Research Thrust for Reliable Plasma Heating and Current Drive using ICRF*
8. D.A. D'Ippolito and J.R. Myra, *ICRF-Edge and Surface Interactions*
9. Edge Coordinating Committee (ECC), *Integrated Edge-Plasma and Plasma-Wall Interaction Research*
10. L. El-Guebaly and R. Raffray, *Impact of Tritium Breeding on Design Implications to Withstand Disruptions and Runaway Electrons*
11. A.M. Garofalo, et al., *Divertor Plans and Research Options in FDF*
12. R. J. Goldston, *The Role of a Long-Pulse, High-Heat-Flux, Hot-Walls Confinement Experiment in the Study of Plasma-Wall Interactions for CTF and Demo*
13. R.J. Goldston, *The Role of Long-Pulse, High-Heat-Flux, Hot-Walls Confinement Experiment in the Development of Plasma Facing Components for CTF and Demo*
14. A. Hassanein, V. Sizyuk, G. Mikoshevsky, S. Harilal, and T. Sizyuk, *Thrust for Enhancing Modeling and Simulation of Plasma Instabilities/Surface Interactions with innovative mitigation techniques*
15. S. Hsu, X. Tang, Y. Wang, and G. Tynan, *Synergistic Effects of Radiation Damage and Plasma-Material Interactions*
16. M. Kotschenreuther, S.M. Mahajan, P. Valanju, J.M. Canik, A.M. Garofalo, B. LaBombard, and R. Maingi, *Severe divertor issues on next step devices, and validating the Super-X divertor as a promising solution*

17. P.S. Krstic, F.W. Meyer, Y.K. -M. Peng, D.L. Hillis, L.R. Baylor, R.H. Goulding, J.H. Harris, *Fundamental science of the synergy of multi-species interactions in a high plasma-heat-flux environment*

18. B. LaBombard, *Towards building a credible vision for a DEMO-class fusion reactor: Addressing the 'Knowledge Gaps' and 'Show-Stoppers' in Edge Plasma Transport*

19. A.W. Leonard, *A Fusion Development Facility to Test Divertor and PFC Solutions for DEMO*

20. A.W. Leonard, S. Allen, T. Petrie and P. Stangeby, *Development and Validation of a Boundary Plasma Model*

21. R.P. Majeski, J.P. Allain, H. Ji, N. Morley, M. Nornberg, and D. Ruzic, *Liquid Metal Plasma-Facing Components*

22. J. Menard, J.-K. Park, and S.A. Sabbagh, *Physics and engineering research needs for 3D coil systems for a tokamak DEMO*

23. N.B. Morley, M. Abdou, A. Ying, M. Sawan, J. Blanchard, C. Wong, B.J. Merrill, and P. Calderoni, *Capitalizing on the ITER Opportunity – an ITER-TBM Experimental Thrust*

24. R.E. Nygren, *Future Plasma Facing Components (PFCs) & In-vessel Components (IVCs): Strengthened Sustained and Integrated Approach for Modeling and Testing HHFC*

25. R.E. Nygren, *In-vessel Engineering Instrumentation for Future Fusion Devices*,

26. W.A. Peeble and J. Irby, *Research Thrust to Address Major Measurement Gaps*

27. PWI Panel, *PWI Gaps vs. Tools: General Requirements on the Tools*

28. T.D. Rognlien, D. Ryutov, M. Makowski, V. Soukhanovski, M. Umansky, R. Cohen, D. Hill and I. Joseph, *Innovative divertor development to solve the plasma heat-flux problem*

29. T.D. Rognlien, *Prediction of PFC plasma fluxes by improved edge/scrape-off-layer simulations*

30. C.H. Skinner, *Management of dust in fusion devices*

31. R.D. Stambaugh, V.S. Chan, A.M. Garofalo, et al., *Research Thrusts Made Possible by a Fusion Development Facility*—executive summary

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33. P. Stangeby, *Carbon as a flow-through, consumable PFC material*

34. P. Stangeby, *Diagnostic investments required to identify missing physics controlling PWI*

35. P. Stangeby, *FDF: PWI issues and research opportunities*

36. P. Stangeby, *The process of identifying the physics controlling PWI is significantly incomplete, preventing reliable scaling of PWI results to future devices*

37. E.J. Strait, J.C. Wesley, M.J. Schaffer, and M.A. Van Zeeland, *Off-Normal Events in a Fusion Development Facility*
38. R.J. Temkin, M.A. Shapiro, J.R. Sirigiri, and C.P. Moeller, *RF Launchers that Survive in the Fusion Reactor Environment*
39. M. Tillack, *Evaluating gaps in fusion energy research using Technology Readiness Levels*
40. D. Whyte, *An Energy Sustainment Science Mission*
41. D. Whyte, *Taming the Plasma-Material Interface and an Energy Sustainment Mission*
42. J.R. Wilson, R.J. Goldston and J.C. Hosea, *The Problem of RF Launchers in a DEMO Environment and Requirements to Address them*
43. C.P.C. Wong, et al., *BW-Surface for CTF and DEMO*
44. G.A. Wurden, *The Real Mission of ITER*
45. D. Youchison, *Plasma Facing Component Test Facilities*
46. D. Youchison, *The Case for Helium-Cooled Refractory PFCs*

THEME 4: HARNESSING FUSION POWER

1. L.R. Baylor, *Fusion Burn Control with Isotopic Fueling*
2. G. Bench and M.J. Fluss, *Triple Ion-Beam Radiation Materials User Facility: A Facility Upgrade of National Importance*
3. M.A.M. Bourke, D.J. Rej, R.D. Fulton, J.L. Sarrao, *Material characterization in a 10^{15} neutron $cm^{-2}s^{-1}$ flux using light source X-rays*
4. T. Burgess, Y.K.-M Peng, and M. Cole, *RAMI Research Thrust Using a Full Fusion Nuclear Environment*
5. R.J. Buttery, *The Need for a Fusion Science Integration Experiment in the US*
6. L. Cadwallader, *Fusion Operating Experience Data Collection at Present and in the Future*
7. J.M. Canik, A. Sontag, M. Cole, M. Kotchenreuther, P. Valanju, *Limiting the Divertor Heat Flux to Enable Fusion Nuclear Science Research at Low-A*
8. M. Cole, Y.K.-M. Peng, B. Nelson, and T. Burgess, *Research Thrust on Magnets to Enable Low Aspect Ratio*
9. S. Diem, D. Batchelor, V. Chan, A.M. Garofalo, Y.K.-M. Peng, D. Spong and A. Sontag, *Research Thrust to Establish Predictive Simulation Capability for Fusion Nuclear Science*
10. L. El-Guebaly and R. Raffray, *Impact of Tritium Breeding on Design Implications to Withstand Disruptions and Runaway Electrons*
11. L. El-Guebaly and S. Malang, *Need for Online Adjustment/Control of Tritium Bred in Blanket*
12. L. El-Guebaly, *Establishing Integrated Management Strategy for Activated Materials*
13. L. El-Guebaly, *Need for High Tritium Burn-up Fraction in Plasma to Relax Tritium Breeding Requirement for Demo and Fusion Power Plants*
14. R.J. Goldston, *Requirements for a Confinement Device with a Goal to Develop Tritium Breeding Blanket Modules, Based on FESAC Fusion Development Path Plan*
15. S. Hsu, X. Tang, Y. Wang, and G. Tynan, *Synergistic Effects of Radiation Damage and Plasma-Material Interactions*
16. T.P. Intrator and R.P Majeski, *Radio frequency sustainment and current drive for Compact Tori*
17. T.P. Intrator, T. Jarboe, and R.L. Miller, *Reactor considerations for Compact Toroids*
18. K. Korsah, B. Patton, and Y.K.-M. Peng, *Safety Research Thrust Using A Full Fusion Nuclear Environment*
19. R.J. Kurtz and G.R. Odette, *Major Materials Issues for DEMO*

20. S.A. Maloy and E. Pitcher, *Fusion Materials Qualification with a Spallation Neutron Source Facility*

21. A. Misra, R.G. Hoagland, M. Nastasi, M.J. Demkowicz,, S.A. Maloy, T.C. Germann, Y.Q. Wang, X.Y. Liu, B.P. Uberuaga, and A.F. Voter, *Atomic-Scale Design of Structural Materials for Fusion Environments*

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25. R.E. Nygren, *In-vessel Engineering Instrumentation for Future Fusion Devices*

26. G.R. Odette and S. Sharafat, *Extreme Thermo-mechanical Challenges In Developing Fusion Power Reactor Materials and Structures*

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29. E. Pitcher and S. Maloy, *Irradiating Fusion Materials in the Proposed LANL Materials Test Station*

30. S. Sharafat and G.R. Odette, *Interfacing Fusion Materials Development and Component Design*

31. J. Sheffield, *Component Test Facilities will be needed*

32. J. Sheffield, *D-D Power Plants: the ultimate fusion goal*

33. T.C. Simonen and the Mirror Study Group, *A Materials Evaluation D-T Neutron Source Based on an Axisymmetric Gas Dynamic Trap Magnetic Mirror*

34. J. Slough, *Development of a high fluence fusion neutron source and component test facility based on the magneto-kinetic compression of FRCs*

35. S. Smolentsev, *Identification of key MHD thermofluid issues and associated R&D for the next generation liquid blankets*

36. A. Sontag, J.B.O. Caughman, S. Diem, R. Fonck, A.M. Garofalo, R. Raman, A. Redd, and V. Shevchenko, *Research Thrust on Plasma Startup & Ramp-up to Enable Fusion Nuclear Science Research at Low-A*

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43. L.M. Waganer, *An Integrated Design for Demo-FPP to Achieve RAMI Goals*
44. J. Wagner, Y.K.-M. Peng, M. Sawan, P. Wilson and A. Ying, *Radiation Transport Modeling & Simulation for a Full Fusion Nuclear Environment*
45. D. Whyte, *An Energy Sustainment Science Mission*
46. C.P.C. Wong, C.C. Baker, and S.L. Milora, *US Fusion DEMO Assessment*
47. C.P.C. Wong, R.D. Stambaugh, and M. Sawan, *Closing the Fusion Fuel Cycle*
48. G.A. Wurden, *The Real Mission of ITER*
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50. G.L. Yoder, Jr. and Y.K.-M. Peng, *Power Extraction Research Thrust Using a Full Fusion Nuclear Environment*

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1. P.M. Bellan, *Improving spheromak helicity injection by eliminating the constraint imposed by Paschen breakdown physics*
2. E.V. Belova, M. Yamada, H. Ji, S. Gerhardt, *Oblate FRC concept with NBI for experimental studies of large-s FRC formation, stability, current drive and transport*
3. A. Boozer, G.H. Neilson, and M. Zarnstorff, *Non-Axisymmetric Shaping as a Research Thrust*
4. R.J. Buttery, *The Need for a Fusion Science Integration Experiment in the US*
5. J.M. Canik, A. Sontag, M. Cole, M. Kotchenreuther, P. Valanju, *Limiting the Divertor Heat Flux to Enable Fusion Nuclear Science Research at Low-A*
6. S.A. Cohen, *Improving energy confinement is the most rapid route to practical FRC reactors*
7. D.J. Den Hartog, *Measurement needs for the alternates, with broader connections*
8. S. Diem, D. Batchelor, V. Chan, A.M. Garofalo, Y.K.-M. Peng, D. Spong and A. Sontag, *Research Thrust to Establish Predictive Simulation Capability for Fusion Nuclear Science*
9. L. El-Guebaly and R. Raffray, *Impact of Tritium Breeding on Design Implications to Withstand Disruptions and Runaway Electrons*
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11. R.J. Goldston, J. Menard, and S.A. Sabbagh, *Contributions of NHTX to the ST Development Path*
12. L.R. Grisham, *Neutral Beam Development Plans and Needs for ITER*
13. A.L. Hoffman, L.C. Steinhauer, H. Ferrari, and R. Farengo, *Advances in Singly-Connected Closed Field Line Plasma Devices and Extrapolation to POP Level Experiments and Reactors*
14. W. Horton, J. Pratt, A.W. Molvik, R.M. Kulsrud, and N.J. Fisch, *Axisymmetric Tandem Mirror D-T Neutron Source*
15. P. Innocente, M.E. Puiatti, and M. Valisa, *Density limits and control in RFP's*
16. T.P. Intrator and R.P. Majeski, *Radio frequency sustainment and current drive for Compact Tori*
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18. I. R. Jones, *Some thoughts on the future direction of rotamak research*
19. M. Kotschenreuther, S. M. Mahajan, P. Valanju, J.M. Canik, A.M. Garofalo, B. LaBombard, and R. Maingi, *Severe divertor issues on next step devices, and validating the Super-X divertor as a promising solution*

20. J. Leuer and D. Humpreys, *Solenoidless Startup Research Thrusts for Tokamaks*

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22. L. Marrelli, *Issues on active control of MHD instabilities in a RFP*

23. S. Masamune, *A scenario for steady state low-aspect-ratio RFP sustained by pressure-driven and RF/NBI-driven currents*

24. H. McLean, C.R. Sovinec, S. Woodruff, and M. Yamada, *Cross-cutting Connections between the Spheromak and the Reversed-Field Pinch (RFP)*

25. R. Moyer, T.E. Evans, R. Groebner, H. Reimerdes, and P. Snyder, *Pedestal Optimization for Confinement and ELM Control*

26. G.H. Neilson and J. Harris, *Advancing Stellarator Knowledge Through International Collaboration*

27. R.E. Nygren, *Future Plasma Facing Components (PFCs) & In-vessel Components (IVCs): Strengthened Sustained and Integrated Approach for Modeling and Testing HHFCs*

28. Y.K.-M. Peng, M. Cole, B. Patton, G. Yoder, K. Korsah, T. Burgess, Y. Katoh, J. Wagner, A. Sontag, J.M. Canik, S. Diem, R.D. Stambaugh, V.S. Chan, C.P.C. Wong, S. Kaye, *Fusion Nuclear Science Research Thrust and the Required Full Fusion Nuclear Environment*

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31. R. Raman, T.R. Jarboe, and H.W. Kugel, *Research Thrust on Advanced Plasma Fuelling*

32. A. Reiman, *Development of a Predictive Simulation Capability for Three-Dimensional Configurations*

33. A. Reiman, *Stellarator Operational Limits*

34. J. Slough, *Development of a high fluence fusion neutron source and component test facility based on the magneto-kinetic compression of FRCs*

35. J. Slough, *Next Step Needs for FRC Stability And Sustainment Studies*

36. A. Sontag, J.B.O. Caughman, S. Diem, R. Fonck, A. Garofalo, R. Raman, A. Redd, and V. Shevchenko, *Research Thrust on Plasma Startup & Ramp-up to Enable Fusion Nuclear Science Research at Low-A*

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38. D.A. Spong, *Fast ion stability and wall heat loading in stellarators*

39. P.W. Terry, A. Boozer, J.S. Sarff, R. White, and C. R. Sovinec, *Role of Theory in Addressing RFP Physics Issues*

40. K. Tritz, D. Stutman, L. Delgado-Aparicio, and M. Finkenthal, *Understanding and Predicting Microtearing Instabilities in the ST*

41. D. Whyte, *White paper on alpha losses and high-Z metal PFC surface damage*

42. S. Woodruff, *Harnessing fusion power: how does CT research contribute to this discussion?*

43. L. E. Zakharov, *LiWall ST1, EAST1, ITER-100 - all exceeding ignition criterion*

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